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December 16, 1982

MEMORANDUM FOR: D. G. Eisenhut, Director, Division of Licensing

FROM: J. L. Crews, Director, Division of Resident, Reactor Projects and Engineering Programs

SUBJECT: APPARENT DEFICIENCIES IN MIDLAND-ROSS "SUPERSTRUT" MATERIAL USED FOR CLASS 1 CABLE TRAY AND CONDUIT SUPPORTS

The purpose of this memo is to forward the following information with our recommendation that appropriate board notification be considered. This matter was discussed by telephone with members of your staff (T. Novak, B. Buckley, and H. Rood) on December 15 and 16, 1982.

Due to allegations regarding inadequacies in materials and welding, an inspector from the NRC Region IV Vendor Program Branch (VPB) conducted an inspection of the Midl[illegible]d-Ross Superstrut manufacturing facility in Oakland, California, during the period December 6-8, 1982. This facility manufactures mild steel fittings, brackets, and channels, some of which are used to construct cable tray, conduit, and instrument supports in nuclear power plants. The Region IV inspector informed the Region V staff of his findings at the Midland-Ross Oakland facility which included: (1) there was no formal Quality Assurance (QA) program prior to 1979, (2) there were no records of the qualification of welding operators or welding procedures, (3) prior to 1980, spot welds were not sample tested and not controlled by procedures, (4) there was no traceability of material, (5) there were no quality records before 1980, and (7) generally, the current QA program did not meet the intent of 10 CFR 50, Appendix B criteria. The VPB inspector also informed the Region V staff that the Superstrut material manufactured at the Oakland facility had been used at nuclear power plants in Region V, including A[illegible]na Public Service's Palo Verde, Pacific Gas and Electric's Diablo Cany[illegible] [illegible]d Washington Public Power Supply System's WNP 1 and 4 plants. The e[illegible]t [illegible] applicability to other NRC licensed facilities is unknown at this [illegible]

Region V dispatched an inspector on December 8, 1982, to conduct a special inspection of one of the affected facilities (Diablo Canyon) to determine the scope and potential impact of the VPB inspector's findings. The Region V inspector found that the back-to-back double channels which were spot welded together, as well as the channels with welded end brackets, were widely used

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(up to 11,000 supports out of approximately 24,000 in the Diablo Canyon facility) and that the licensee's engineering staff had treated the double channel Superstrut material as a composite member and not as two members acting independently. The Region V staff has alerted appropriate NRR staff personnel regarding the situation described above and is preparing a special inspection report on their inspection findings.

Should you require additional information regarding this subject, please do not hesitate to call me (FTS 463-3735) or Phil Morrill (FTS 463-3740).

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Distribution for Board Notification (83-02)

Document Control (50-483, 50-461/462, 50-445/446, 50-275/323, 50-329/330, 50-528/529, 50-322, 50-498/499, 50-382, 50-358, 50-341, 50-361/362, 50-395, 50-437)

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 3 0 1982

MEMORANDUM FOR: The Atomic Safety & Licensing Boards for:

Callaway Plant, Unit 1
Clinton Power Station, Units 1/2
Comanche Peak Steam Electric Station, Units 1/2
Midland Plant, Units 1/2
South Texas Project 1/2
Waterford Steam Electric Station, Unit 3
William H. Zimmer Nuclear Power Station, Unit 1, and

The Atomic Safety & Licensing Appeal Boards for:

Comanche Peak Steam Electric Station, Units 1/2
Enrico Fermi Atomic Power Plant, Unit 2
Susquehanna Steam Electric Station, Units 1/2
Waterford Steam Electric Station, Unit 3
William H. Zimmer Nuclear Power Station, Unit 1

FROM: Thomas M. Novak, Assistant Director for Licensing
Division of Licensing, NRR

SUBJECT: BOARD NOTIFICATION - USGS POSITION ON THE CHARLESTON EARTHQUAKE (Board Notification 82-122A)

We have recently received the enclosed letter from the U. S. Geological Survey (USGS) (Letter, James F. Devine to Robert E. Jackson, November 18, 1982) which clarifies previous recommendations made by the USGS to NRC regarding the reoccurrence of the 1886 Charleston-type earthquake. This clarification has been provided after lengthy deliberations by the USGS. The possibility of this clarification was identified in SECY-82-53.

For the purpose of licensing of facilities in the Southeastern U. S., the NRC has taken a position, based primarily on the advice of the U. S. Geological Survey (USGS), that any reoccurrence of the 1886 Charleston, S.C. earthquake (Modified Mercalli Intensity (MMI) X, estimated Magnitude about 7) would be confined to the Charleston area. That is, the Charleston earthquake is assumed to be associated with a geologic structure in the Charleston area. Nuclear power plants in the region east of the Appalachian Mountains are, therefore, usually controlled in their seismic design,

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we are transmitting it to boards for all plants east of the Rocky Mountains. Since some question may exist regarding its technical applicability, our evaluation of the significance of this clarification is underway. We will inform the appropriate Boards regarding any significant changes in the staff's position as a result of the evaluation.

O. Schwencer / for

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: Licensee/Boards
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UNITED ST[illegible] AMERICA
NUCLEAR RE[illegible] COMMISSION

BEFORE THE ATOMIC S[illegible] AND LICENSING BOARD

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|---|---|---|
| In the Matter of | ) | |
| | ) | |
| UNION ELECTRIC COMPANY | ) | Docket No. STN 50-483 OL |
| | ) | |
| (Callaway Plant, Unit 1 | ) | |

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NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

| | | |
|---|---|---|
| In the Matter of | ) | |
| | ) | |
| ILLINOIS POWER COMPANY, et al. | ) | Docket No. 50-461 OL |
| | ) | |
| (Clinton Power Station, Unit 1) | ) | |
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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|---|---|---|
| In the Matter of<br><br>TEXAS UTILITIES GENERATING COMPANY, et al.<br><br>(Comanche Peak Steam Electric Station, Units 1 and 2) | ) | Docket Nos. 50-445<br>50-446 |

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I hereby certify that copies of "NRC STAFF ANSWER TO CASE MOTIONS SEEKING ADMISSION OF DOCUMENTS" in the above captioned proceeding have been served on the following by deposit in the United States mail, first class or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, this 4th day of November, 1982.

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NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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|---|---|---|
| In the Matter of | ) | |
| | ) | |
| CONSUMERS POWER COMPANY | ) | Docket Nos. 50-329 OM & OL |
| | ) | 50-330 OM & OL |
| (Midland Plant, Units 1 and 2) | ) | |

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BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

| | | |
|---|---|---|
| In the Matter of | ) | |
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| | ) | |
| (Enrico Fermi Atomic Power Plant, Unit 2) | ) | |

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| In the Matter of | ) | |
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| HOUSTON LIGHTING AND POWER COMPANY, ET AL. | ) | Docket Nos. 50-498 50-499 |
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|---|---|---|
| In the Matter of | ) | |
| | ) | |
| PENNSYLVANIA POWER AND LIGHT CO. | ) | Docket Nos. 50-387 |
| ALLEGHENY ELECTRIC COOPERATIVE, INC. | ) | 50-388 |
| | ) | |
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| Units 1 and 2) | ) | |

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

| | | |
|---|---|---|
| In the Matter of | ) | |
| | ) | |
| LOUISIANA POWER AND LIGHT COMPANY | ) | Docket No. 50-382 |
| | ) | |
| (Waterford Steam Electric Station, Unit 3) | ) | |

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

| | | |
|---|---|---|
| In the Matter of | ) | |
| | ) | |
| CINCINNATI GAS AND ELECTRIC | ) | Docket No. 50-358 |
| COMPANY, et al. | ) | |
| | ) | |
| (Wm. H. Zimmer Nuclear Power | ) | |
| Station, Unit No. 1) | ) | |

## CERTIFICATE OF SERVICE

I hereby certify that coies of "NRC STAFF RESPONSE TO APPLICANTS' BRIEF IN SUPPORT OF THEIR REVISED EXCEPTIONS" ERRATA in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, this 15th day of December, 1982:

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Atomic Safety and Licensing Appeal Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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United States Department of the Interior

GEOLOGICAL SURVEY
RESTON, VA. 22092

In Reply Refer To:
Mail Stop 905

NOV 1 8 1982

Dr. Robert E. Jackson
Chief, Geosciences Branch
Division of Engineering
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Bob:

The purpose of this letter is to clarify our position on the seismic potential of certain regions of the Eastern United States. In our letter of December 30, 1980, on the same subject we expressed the view that ". . . the likelihood of a Charleston sized event in other parts of the Coastal Plain and Piedmont is very low."

As you are aware, after several years of intensive study in the Charleston region, no geologic structure or feature can be identified unequivocally as the source of the 1886 Charleston earthquake. However, as studies in the Charleston region and elsewhere along the Atlantic margin have progressed, it has become evident that the general geologic structure of the Charleston region can be found at other locales within the eastern seaboard (Appalachian Piedmont, Atlantic Coastal Plain, and Atlantic Continental Shelf).

Because the geologic and tectonic features of the Charleston region are similar to those in other regions of the eastern seaboard, we conclude that although there is no recent or historical evidence that other regions have experienced strong earthquakes, the historical record is not, of itself, sufficient grounds for ruling out the occurrence in these other regions of strong seismic ground motions similar to those experienced near Charleston in 1886. Although the probability of strong ground motion due to an earthquake in any given year at a particular location in the eastern seaboard may be very low, deterministic and probabilistic evaluations of the seismic hazard should be made for individual sites in the eastern seaboard to establish the seismic engineering parameters for critical facilities.

As stated in our letter of December 30, 1980, earthquakes similar to the 1886 Charleston, South Carolina, event should be considered as having the potential to occur in the vicinity of Charleston and seismic engineering parameters of critical facilities in that area should be determined on that basis.

Sincerely yours,

James F. Devine

James F. Devine
Assistant Director for
Engineering Geology

8212020193

November 19, 1982

FOR: The Commissioners

FROM: Executive Director for Operations

SUBJECT: CLARIFICATION OF U. S. GEOLOGICAL SURVEY POSITION RELATING TO SEISMIC DESIGN EARTHQUAKES IN THE EASTERN SEABOARD OF THE UNITED STATES

PURPOSE: To provide the Commissioners with information relating to the clarification of the U. S. Geological Survey Position with respect to the 1886 Charleston, S.C. Earthquake reoccurrence

DISCUSSION For the purpose of licensing of facilities in the Southeastern U. S., the NRC has taken a position, based primarily on the advice of the U.S. Geological Survey (USGS), that any reoccurrence of the 1886 Charleston, S.C. earthquake (Modified Mercalli Intensity (MMI) X, estimated Magnitude about 7) would be confined to the Charleston area. That is, the Charleston earthquake is assumed to be associated with a geologic structure in the Charleston area. Nuclear power plants in the region east of the Appalachian Mountains are, therefore, usually controlled in their seismic design, according to Appendix A to 10 CFR Part 100, by the maximum historical earthquake not associated with a geologic structure. This controlling earthquake is typically an MMI VII or VIII. Since 1974, the NRC has funded an extensive research project in the Charleston area to gain further information on the causative mechanism of this event.

On January 28 and 29, 1982 the Extreme External Phenomenona Subcommittee of the ACRS convened a meeting of expert professionals in the geosciences to obtain an overview of the state of knowledge and future NRC research needs in this area. During that meeting, we were informed by the USGS that it had formed a working group to reassess the validity of its position on the Charleston earthquake.

8212130064

Contact:
R. Vollmer, NRR
492-7207

Our evaluation of the significance of this clarification is underway. Currently, a two day review meeting between NRC (ORES and ONRR) and the USGS is planned for November 30, 1982 and December 1, 1982 to discuss both the status of geoscience knowledge in the Charleston region and future research efforts. The first day will be an open public meeting (noticed in the Federal Register) which will allow for comments and questions from interested parties and members of the public.

We have also attached our preliminary views on a plan to address this clarified USGS position. This plan includes elements which relate to both ongoing research and licensing efforts and possible requirements for new efforts (split approximately 75% and 25% respectively). This plan will be modified and completed after several meetings with the USGS take place in order that a more complete understanding of its clarified position can be obtained.

William J. Dircks
Executive Director for Operations

Attachments:
As stated

United States Department of the Interior

GEOLOGICAL SURVEY
RESTON, VA. 22092

In Reply Refer To:
Mail Stop 905

NOV 18 1982

Dr. Robert E. Jackson
Chief, Geosciences Branch
Division of Engineering
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Bob:

The purpose of this letter is to clarify our position on the seismic potential of certain regions of the Eastern United States. In our letter of December 30, 1980, on the same subject we expressed the view that ". . . the likelihood of a Charleston sized event in other parts of the Coastal Plain and Piedmont is very low."

As you are aware, after several years of intensive study in the Charleston region, no geologic structure or feature can be identified unequivocally as the source of the 1886 Charleston earthquake. However, as studies in the Charleston region and elsewhere along the Atlantic margin have progressed, it has become evident that the general geologic structure of the Charleston region can be found at other locales within the eastern seaboard (Appalachian Piedmont, Atlantic Coastal Plain, and Atlantic Continental Shelf).

Because the geologic and tectonic features of the Charleston region are similar to those in other regions of the eastern seaboard, we conclude that although there is no recent or historical evidence that other regions have experienced strong earthquakes, the historical record is not, of itself, sufficient grounds for ruling out the occurrence in these other regions of strong seismic ground motions similar to those experienced near Charleston in 1886. Although the probability of strong ground motion due to an earthquake in any given year at a particular location in the eastern seaboard may be very low, deterministic and probabilistic evaluations of the seismic hazard should be made for individual sites in the eastern seaboard to establish the seismic engineering parameters for critical facilities.

As stated in our letter of December 30, 1980, earthquakes similar to the 1886 Charleston, South Carolina, event should be considered as having the potential to occur in the vicinity of Charleston and seismic engineering parameters of critical facilities in that area should be determined on that basis.

Sincerely yours,

James F. Devine

James F. Devine
Assistant Director for
Engineering Geology

~~8212020193~~

Outline for Recommended Plan
Eastern U. S. Earthquakes

## Introduction

Based on our preliminary assessment of the U. S. Geological Survey's (USGS) clarification of position relating to a Charleston-type earthquake, we do not see a need for any immediate action for specific sites at this time. Instead, we foresee that this clarification can be addressed predominantly through existing ongoing programs at NRC with the possibility of additional requirements for work by the Utilities.

The USGS clarification indicates that deterministic and probabilistic evaluations should be made. Generally, for most existing sites, extensive deterministic studies have been undertaken and used in developing the existing seismic design basis. We therefore believe that this element of the clarification continue to be addressed through our long range research plan. Specific modifications to that plan can be made in order to address specific tectonic structures. If necessary, a few specific applicants or licensees may be required to investigate tectonic structures which may not have been previously identified during the licensing procedure.

As many of the current working deterministic hypotheses are not directly amenable to investigation in the short term, we believe that the clarification issue should be pursued in the short term principally through a probabilistic assessment of plants in the eastern seaboard. This probabilistic program can be coupled to the current ongoing NRC efforts in this area already underway. We also believe that utility-sponsored studies should be undertaken, preferably as a consolidated group, to assess the seismic hazard in the eastern seaboard.

Further specifics on this program will be provided after more extensive discussions with the USGS.

## PROBABILISTIC EVALUATION:

In our view, the USGS clarification represents not so much a new understanding but rather a more explicit recognition of existing uncertainties with respect to the causative structure and mechanism of the 1886 Charleston earthquake. Many hypotheses have been proposed as to the locale in the eastern seaboard of future Charleston-size earthquakes. Some of these could be very restrictive in location while others would allow this earthquake to reoccur over very large areas. Presently, none of these hypotheses are definitive and all contain a strong element of speculation.

Traditional deterministic approaches are not generally designed to deal with this situation. Probabilistic methods which allow for the consideration of many hypotheses, their associated credibilities, and the explicit incorporation of uncertainty are much better equipped to provide rational frameworks for decision making. We believe that the

2. The determination of the geometry of structure and tectonics of the earth's crust at depths where earthquakes are occurring (5-20 km) in the eastern seaboard using such techniques as seismic reflection profiling.

3. The continuation of subsurface neotectonic investigations of earthquake source areas to determine if uplift, subsidence or differential movement is occurring. Such studies may include among others:

   A. Tectonic Geomorphology
   B. Geodetic Measurements
   C. Geologic Mapping
   D. Remote Sensing

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 3 1 1983

MEMORANDUM FOR: Franklin D. Coffman, Jr., Leader
Systems Interaction Section
Reliability and Risk Assessment Branch

FROM: James H. Conran, Senior Systems Engineer
Systems Interaction Section, RRAB

SUBJECT: DIFFERING PROFESSIONAL OPINION

The purpose of this memorandum is to submit formally, in accordance with NRR Manual Chapter 4125, a statement of differing professional opinion regarding certain aspects of existing policy and practice in the areas of systems interaction and safety classification. Many, but not all, aspects of the matters at issue herein were addressed by me earlier in an affidavit dated February 9, 1983, to the Shoreham Hearing Board.

Enclosure 1 to this memorandum sets forth the detailed statement of my differing professional opinion in the areas identified above in the format suggested in Section C of NRC Appendix 4125. In order to avoid needless repetition therein of the detailed treatment given already in the earlier affidavit to matters also of concern in the immediate context, Enclosure 1 draws to the maximum extent possible on the presentation of issues provided in the affidavit. Accordingly, the earlier affidavit is incorporated into this differing professional opinion as Appendix A; and Appendix A and Enclosure 1 are appropriately cross- referenced in order to facilitate their use together. Points addressed to the attention of NRC management in the immediate context that were not treated explicitly in the affidavit to the Shoreham Board are denoted by asterisks in Enclosure 1. Minor changes and editorial-type corrections made to the earlier affidavit since it was executed on 2/9/83 are indicated by a bar in the right margin.

James H. Conran

James H. Conran
Senior Systems Engineer
Systems Interaction Section, RRAB

Attachements: See next page

~~8304080379 XA~~

Attachments:

Enclosure 1. Detailed Statement of Differing Professional Opinion.

Enclosure 2. Excerpt from Statement of Staff Views to Shoreham Board, dated 2/22/83.

Enclosure 3. Memo, dated 3/9/79, Rubinstein to Bradford, "Probabilities That the Next Major Accident Occurs Within Prescribed Intervals".

Enclosure 4. Technical Paper, by D. Rubinstein, dated 2/4/81, "A Statisticians View of NRC Statistics".

Enclosure 5. Technical Paper (Draft), by D. Rubinstein, dated 10/26/81, "Random Thoughts on Uncertainties, Risk Analysis, and Nuclear Regulation".

Enclosure 6. Note, dated 3/18/82, Conran to Coffman, "Comments on Draft Letter (Hanauer to Cooper -NUPPSCO)....and Related Matters".

Enclosure 7. Excerpt from Rebuttal Testimony on Contention 7B, dated 7/1/82, by J. H. Conran.

Appendix A - Affidavit of James H. Conran, dated 2/9/83, to the Shoreham ASLB.

cc: R. J. Rawson, ELD
w/Attachments

Enclosure I

## STATEMENT OF DIFFERING PROFESSIONAL OPINION

### I. Systems Interaction Topic

Issue A. Significant Extension of Schedule for Resolution of USI A-17 Without Appropriate Review or Justification

1. Management View or Position

The staff's program for resolution of USI A-17 has failed to achieve resolution of the systems interaction issue by now, in accordance with the schedule established as reasonable and acceptable when the program was initiated. Notwithstanding, management considers the program to be progressing satisfactorily and emphasizes at this point the "confirmatory" nature of the program.(1) (2) Accordingly:

a. management proposes at this point to continue pursuit of resolution of USI A-17 by following basically the same

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(1) See Statement of Staff Views to the Shoreham Hearing Board, dated 2/22/83. (Excerpt attached as Enclosure 2.)

(2) Also see NRC Staff Supplemental Testimony on Contention 7B (Shoreham OL Proceeding), dated 3/10/83, at p. 5 & p. 14.

approach and program plan employed to date, despite failure of that approach to achieve resolution of the important safety issue involved in the time allotted, and

b. management proposes to simply slip again, significantly, the schedule for resolution of USI A-17, without proper review and consideration of the possible need to accelerate the resolution of this issue and the possible consequences of failing to do so.

SEE APPENDIX A, AT P. 4 AND P. 10-11

2. <u>Differing Professional Opinion</u>

a. Systems interaction in nuclear power plants is designated as both a Priority Category "A" generic safety concern and an Unresolved Safety Issue (i.e., USI A-17). As such, by NRC policy and the agency's own definitions, the issue involved:

- is a matter that poses <u>important</u> <u>questions</u> regarding adequacy of existing requirements, for which resolution is judged necessary to provide a potentially significant decrease in the risk to public health and safety, and whose resolution is likely to result in NRC action,[3] and

o involves a generic concern judged by the staff to warrant priority attention in terms of manpower and funds, that should be pursued promptly to obtain early resolution that could provide possible significant increase in assurance of public health and safety.[4] [5]

b. In view of the importance ascribed to the systems interaction issue and the indicated need for prompt treatment and early resolution (as seen from A.2.a above), failure to achieve resolution of USI A-17 within the period established (by consensus) as acceptable should be treated as an important safety issue in itself. Accordingly, the decision regarding schedule and approach to be followed from this point for resolution of USI A-17 should be made only after

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(3) See NUREG-0510 at p. 10

(4) See NUREG-0510 at p. 11 and p. 49 (Table 1 - Priority Category "A" Definition).

(5) See NUREG-0885 at p. 5 (Commission 1983 Policy and Planning Guidance).

full review and appropriate consideration of the possible safety implications involved in further delay.

SEE APPENDIX A, AT P. 3-9, AND P. 11-12.

c. To assure that there is not further significant delay in resolving the systems interaction issue, an augmented and accelerated approach should be adopted at this point for resolution of USI A-17. Efforts should continue as planned under the current program, for development and demonstration by the staff of improved, efficient methods for comprehensive broad-scope systems interaction evaluations (for later application in all facilities, if found necessary). Additionally, however, all licensee and NTOL applicants should be required to perform <u>limited</u> systems interaction evaluations of their facilities (scope to be established by agreement with staff) using currently-available techniques. This would better ensure early availability of actual in-plant systems interaction data required by the staff to determine the need for full-scope systems interaction studies generically.

This approach would make the program less vulnerable to significant delays resulting from plant-specific operating problems and licensing-related difficulties (as has occurred repeatedly under the current approach), because availability of the required data would no longer be dependent upon completion of studies in just a <u>few</u> "participating" facilities. At the same time, utilities would not be unduly burdened by an immediate requirement for full-scope, comprehensive systems interaction studies that might not be justified at this time.

*d. In the absence of compelling current indication that the definitions and policy indicated in A.2.a above (regarding the nature of items designated Unresolved Safety Issue and/or Priority Category "A") no longer apply to the systems interaction concern, management should not now characterize USI A-17 as merely or principally "confirmatory" in nature.

3. Possible Consequences if Differing Professional Opinion is Not Adopted

If the underlying causes of unexpected events in reactor operating experience, such as common cause/common mode failure, are not addressed effectively (e.g., by timely resolution of USI A-17), the likelihood of a serious accident occurring could become unaccpetably high.

SEE APPENDIX A, AT P. 6-7.

(See also A.4.b below for a more quantitative approach to treatment of the stated concern)

4. Related Efforts and Other Information Pertinent to Resolution of Differing Professional Opinion

a. The ACRS has considered the systems interaction issue in the broad licensing context since 1974, and has made specific recommendations on several occasions regarding the kinds of less-than-full-scope systems interaction evaluations that could be usefully undertaken in both operating plants and NTOL facilities. (See, for example ACRS letters dated 1/8/82 and 3/9/82, regarding systems interaction matters.) The ACRS should be consulted in deciding finally the course of action to be taken from this point in pursuing the systems interaction issue.

*b. Mr. David Rubinstein, a statistician and member of the RRAB staff, has described previously (in a separate context) the "prediction interval method" for putting an upper bound on the probability that (given "X" number of reactor years of operation without a major accident) the <u>next</u> major accident will occur within a specified number of years.[6] That statistical method provides an alternate way of addressing the concern expressed qualtitatively in A.3 above regarding urgency of timely resolution of USI A-17; and it could provide another useful perspective and additional insights in the difficult process of developing a consensus judgment

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[6] See memo, dated 3/9/79, Rubinstein to Bradford, "Probabilities That the Next Major Accident Occurs Within Prescribed Intervals" (See attached, Enclosure 3).

now regarding the general question of acceptable schedule and proper course to be followed from this point in resolving USI A-17, and, more specifically, regarding whether or not an accelerated approach should be taken now in that regard.

Issue B. Disproportionate Emphasis and Priority Given PRA to The Detriment of Systems Interaction Program

1. Management Policy or Practice

There have been clear indications over the last ~2 years of significant decrease in emphasis by NRC management on systems interaction as a licensing-related safety issue requiring early resolution and warranting priority attention in its own right. Concurrently, increased emphasis and high priority has been given to PRA-related programs/activities that are only of a development nature. Examples or manifestations of management attitude and practice in this regard include:

a. abolishment of the Systems Interaction Branch in early 1981, and an accompanying sharp reduction in the number of NRC technical staff assigned to systems interaction efforts within NRR (PRA programs and activities within NRR were not similarly affected),

SEE APPENDIX A, AT P. 16-17.

*b. assignment currently of significantly greater numbers of NRC technical staff (either full time or part time) in support of PRA-related development type programs and development activities than are assigned to the licensing-related USI A-17 effort.

*c. completion, or near-completion, to date of ~15 or more broad-scope PRA studies at reactor facilities under NRC cognizance (including both operating reactors and NTOL plants), whereas not one broad-scope systems interaction study planned in connection with USI A-17 has yet been completed at any facility.

d. withholding/delay (from October 1981 to present) of NRR approval for implementation of the important methodology demonstration phase of the systems interaction program because of (i) cost-benefit concerns, and (ii) lack of any showing that significant "risk-benefit" was to be gained from the systems interaction studies planned in pursuit of resolution of USI A-17.

SEE APPENDIX A, AT P. 19-21, and P. 24.

e. protracted effort (from ~October 1981 to present) to merge the systems interaction program with the NREP program for cost-benefit advantage, without regard to adverse effect on the more important licensing-related objective ( i.e., resolution of USI A-17).

SEE APPENDIX A, AT P. 16-21.

f. progressive blurring of the distinction between systems interaction (a licensing-related USI) and PRA (a developmental-type activity), and a growing tendency to treat systems interaction as <u>just</u> a subordinate part of PRA.

SEE APPENDIX A, AT P. 22-24.

2. <u>Differing Professional Opinion</u>

a. Under the current system of NRC rules and procedures for reactor licensing, the systems interaction issue (i.e., USI A-17) is a matter that <u>must</u> be addressed in determining compliance with existing rules to assure adequate safety. The same cannot be said regarding NRC's PRA-related programs and acitivities. That is an important distinction that should be taken into account and weighted more heavily in determining the relative importance and priorities of systems interaction and PRA-related programs. NRC should continue to pursue PRA-related development programs intended to improve understanding of the risks associated with operation of reactors. Disproportionate emphasis and priority has been given to PRA, however, in the last ~2 years by NRC management; and this has operated to the serious detriment of the systems interaction program, and resulted in inordinate delay in the resolution of USI A-17.

SEE APPENDIX A, AT P. 16-18.

*b. Greater emphasis should be given (e.g., in the Safety Goal Implementation Plan) to the fact that incomplete treatment of systems interaction is a major potential source of uncertainty in PRA, and that further study of the possible need for significant improvement in that regard (e.g., as planned in the USI A-17 program) must be completed before final consideration will be given to approval for use of PRA in licensing applications currently proscribed.

*c. Proper balance should be restored with respect to importance ascribed and priorities given to systems interaction and PRA-related programs by NRC management, reflecting consideration of the important distinction between those two major areas of activity, as indicated in B.2.a above. Specifically, NRC management should:

- o assign higher priority than is currently given to programs for resolution of the systems interaction issue, and provide increased management support and attention to assure expedited treatment and early resolution of the important licensing-related safety issue involved,
- o assign greater numbers of NRC technical staff to systems interaction work (e.g., comparable to staffing levels dedicated to systems interaction work prior to April 1981),
- o review the effectiveness of the current organizational setup within NRR for conduct of systems interaction programs (e.g., consider seriously a return to the organizational structure and alignments in effect at the outset of the II.C.3 program).

SEE APPENDIX A, AT P. 8.

*d. Schedules for performance of programs for resolution of USI A-17 should be established and implemented so as not to be dependent upon, or subordinate to, PRA-related program schedules in any way that would delay achievement of necessary USI A-17 program objectives.

SEE APPENDIX A, AT P. 8 AND P. 21.

*e. Requirements for cost-benefit analyses should not be imposed (or applied) in a way that delays excessively, or interferes with prohibitively, the conduct and timely completion of programs for resolution of Unresolved Safety Issues (in this instance, USI A-17).

SEE APPENDIX A, AT P. 21-22.

f. Estimates of risk-benefit to be gained from doing comprehensive systems interaction analyses, based solely on extrapolations of current PRA results/data, cannot be regarded as accurate or dependable. Since that is the only basis currently for such estimates, risk-benefit should not be used at this time as a decision criterion by management in determining whether or not to approve systems interaction studies proposed in connection with the USI A-17 program.

SEE APPENDIX A, AT P. 24-26.

*g. Lack of effective communication of systems interaction information and perspectives, to all levels of management and to all cognizant staff (both intra-and-inter office) may have been an a factor in the development of the conditions described in preceding Sections A.1 and B.1. Measures should be taken to assure proper flow of communications in that regard, and also to assure dissemination of alternative views regarding the state of development and usefulness of both PRA and systems interaction analysis methods and techniques. Neither are so highly-developed or refined that both cannot continue to profit from the free exchange of the full range of views on the important matters involved.

3. <u>Possible Consequences if Differing Professional Opinion is Not Adopted</u>

*a. If proper balance is not restored with respect to importance ascribed and priorities assigned to systems interaction and PRA-related programs, and if other specific corrective measures are not implemented as indicated in Section A.2 and B.2 above, further inordinate delay in the resolution of USI A-17 will likely result (with possible increased likelihood of serious accident).

SEE APPENDIX A, AT P. 6-7.

*b. If, in advance of resolution of USI A-17, NRC management continues to encourage initiation and performance of current-state-of-the-art PRAs (i.e., without comprehensive systems interaction analyses as an integral part), unnecessary and excessive costs may result for the licensees or applicants involved when/if the performance of separate comprehensive systems interaction analyses (and integration of PRA and systems interaction results) later become necessary (as has happened to PASNY in the case of Indian Point-3).

(See sections 4.c and 4.d below for further development of the point addressed here.)

4. <u>Related Efforts or Other Information Pertinent to Resolution of Differing Professional Opinions</u>

*a. Comments offered by the ACRS and individual ACRS members (in the context of review of Safety Goals Policy Statement, Safety Goal Implementation Plan, and Severe Accidents Policy Statement), (') regarding treatment in PRA of uncertainties due to systems interactions and premature acceptance/use of current PRA methods and results in licensing, should be given further consideration in the light of all the preceding. The Committee should be consulted in resolving this differing professional opinion.

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(') See ACRS letters dated June 9, 1982; September 15, 1982; September ~~28~~ 14, 1982; and January 10, 1982.

*b. Alternative views expressed earlier and separately by Mr. David Rubinstein, RRAB regarding quality or adequacy of current treatment of uncertainties in PRA, and uncritical acceptance of current PRA results[8],[9] are pertinent and should be considered in the resolution of this differing professional opinion.

*c. Preliminary indications from work being done currently at the Indian Point-3 facility are that great effort and expense will be required to fully factor the results of a broad-scope systems interaction study for a given facility into a full-scope PRA for the same facility, where those two efforts have been conducted as separate activities (as at Indian Point-3).

*d. Information submitted recently to the staff on the Indian Point-3 docket indicates that the findings from comprehensive systems interaction analyses may affect significantly the results obtained from current-state-of-the-art PRAs. Results obtained from the systems interaction evaluation of the Indian Point-3 AFW system,

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(8) Paper dated 2/4/81, "A Statisticians View of NRC Statistics". (See attached, Enclosure 4.)

(9) Paper, dated 10/26/81, "Random Thoughts on Uncertainties, Risk Analysis and Nuclear Regulation," (See attached, Enclosure 5.)

when factored into the Indian Point-3 PRA, nearly doubled the system failure rate for AFW (even after modifications were made to the plant to improve/remove interactions identified).[10]

Corresponding seismic core melt frequency was not found to change appreciably for the case recomputed; but it should be noted that systems interaction search results for other IP-3 systems (in particular, systems that provide alternate cooling in the event of AFW system failure) are only now being separately evaluated in the final phase of the systems interaction analysis effort, and were not factored into the recomputation of IP-3 PRA results that was done at this time.

Also, core melt frequency was not recomputed at this time for the case in which the IP-3 PRA model was modified to include the AFW systems interactions, but fixes were not made to the plant to remove/improve interactions found. (That case would clearly provide the better comparison and more accurate measure of the full impact i.e., "risk-benefit, of systems interaction analyses on PRA results.)

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[10] See PASNY submittal dated 2/7/83, at p. 4-14 of Attachment II.

## II. Safety Classification Topic

### Issue C. Insufficient Priority Given to Resolution of Known Safety Classification Problems

1. Management Position or Practice

*a. The use of the safety classification terms "safety-related", "safety-grade", and "important to safety" inconsistently or interchangeably by individual staff members was recognized as a problem by NRR management ~2-3 years ago. The immediate problem was dealt with effectively by issuance of guidance to the NRR staff in the form of "standard definitions" for the terms involved (derived directly from the language of the regulations themselves).(12) NRR management has not acted expeditiously, however, in following up that action with additional remedial measures that were also prudently indicated, and which were recommended specifically,(13) i.e.:

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(12) See Memos, dated 11/20/81, Denton to All NRR Personnel and Denton to Mattson, Eisenhut, Vollmer, et al, "Standard Definitions for Commonly-Used Safety Classification Terms.

(13) See Note, dated 3/18/82, Conran to Coffman, "Comments on Draft Letter Hanauer to Cooper -NUPPSCO)....and Related Matters"; and attached routing slip. (See attached - Enclosure 6).

- modification of the regulations to more clearly delineate for all interested and affected parties the general safety classification definitions that are already included (diffusely or reconditely) in the language of the regulations
- development of more formal and detailed guidance (e.g., Reg. Guides or SRP sections) for use by licensees/applicants and all NRC staff (not just NRR), in applying these terms correctly in specific design and licensing review applications.

*b. Reasons given for not pursuing more vigorously the followup measures indicated and recommended were:

- resource availability problems, given the magnitude of the (projected) effort to develop/issue formal guidance documents,
- NRR guidance, although not distributed officially outside NRC (and not binding in present form even if distributed) has been circulated widely (albeit informally) outside the agency so the staff's view/position with regard to definition for the terms involved is widely-known anyway,
- no safety problems or serious potential safety problems are known to have resulted from lack of the more formal and detailed guidance recommended,

- the "problem" involved was thought to be simply or chiefly a "language" problem (i.e., resulting simply from inconsistent or mistaken usage of words applied in treating or discussing safety classification concepts embodied in the regulations that are for the most part mutually understood and agreed upon.

2. Differing Professional Opinion

a. Testimony developed recently in the discussion of safety classification issues in the Shoreham hearing indicates clearly now that lack of unambiguous, detailed guidance regarding the definition and proper application of the classification term "important to safety" can lead to confusion and misunderstanding with respect to the intent of the regulations, and to the development of circumstances that appear to have significant potential adverse safety implications. Specifically, in the Shoreham case cited the applicant has interpreted the term "important to safety" to be equivalent to the term "safety-related" (as both the staff and the applicant understand the term "safety-related"), and has applied that interpretation throughout the design and construction of their facility.

Under this interpretation the applicant, in effect, does not acknowledge any requirements under the regulations for plant features designated by the staff "important to safety, but not safety-related." Said another way, the minimum set of safety requirements recognized by the applicant under this interpretation is considerably smaller than the minimum set of safety requirements recognized by the staff. Such a fundamental difference of understanding regarding what is required <u>minimally</u> by the regulations for adequate safety clearly has significant potential for adverse safety impact. The full implications of the situation indicated in the preceding (particularly in the context of <u>operating</u> facilities) has not yet been completely sorted out;[(14)] but NRC should give high priority now to an effort to do that. At a minimum the measures recommended below should be implemented in remedy of the situation indicated.

SEE APPENDIX A, AT P. 30-33.

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(14) See Rebuttal Testimony dated 7/1/82 by J. H. Conran to the Shoreham ASLB, at p. 6-7. Enclosure 7

*c. NRC should give higher priority now to implementing additional (followup) measures recommended previously, but not yet acted upon, as indicated in C.1.b above.

*d. NRC should complete expeditiously now efforts already initiated for development of a listing of structures, systems, and components "important to safety, but not safety-related" (analogous to the listing of safety-related things in Reg. Guide 1.29), to facilitate proper understanding and application of the intent of the regulations by those who have not previously understood and applied the term "important to safety" in the same way as the staff.

*e. NRC should give high priority now to completion of the joint effort initiated in September 1982 by the staff and industry to develop a safety classification standard for endorsement finally by the staff in a Reg. Guide. This has never been done, and has contributed to the persistence of this problem for many years.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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In the Matter of

LONG ISLAND LIGHTING COMPANY
(Shoreham Nuclear Power Station,
Unit 1)

Docket No. 50-322(OL)

................................

## AFFIDAVIT OF JAMES H. CONRAN

I, James H. Conran, being duly sworn, depose and state that:

## QUALIFICATION OF .'ITNESS

1. I am an employee of the U. S. Nuclear Regulatory Commission (NRC). My present position is Senior Systems Engineer, Reliability and Risk Assessment Branch, Division of Safety Technology within the Office of Nuclear Reactor Regulation. A copy of my professional qualifications is bound into the transcript of the Shoreham Hearing at p. 6538.

APPENDIX A

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## PURPOSE OF AFFIDAVIT

1. The purpose of this affidavit is to identify for the Board (1) areas in which I believe that testimony which I provided earlier in the litigation of Contention 7B requires (or may require) amending and/or supplementing, and (2) changes that have occurred in facts or circumstances material to the matters at issue in Contention 7B which give rise to the need for amending and/or supplementing the testimony involved. The affected testimony falls into two general topic areas, systems interaction and safety classification.

## SYSTEMS INTERACTION TOPIC

2. Change to Testimony and General Circumstance Dictating Change

Consistent with the Appeal Board's decision in North Anna[1], staff's testimony on systems interaction in the Shoreham hearing included a discussion of Unresolved Safety Issue A-17, with the specific objective of demonstrating "justification for operation" of Shoreham despite pendency of that USI. I was the principal author of the portion of staff's written testimony covering systems interaction, and was a principal witness in presenting the staff's position on that issue before the Board. My testimony in that regard was based necessarily on my understanding, at the times that that testimony was written

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[1] See ALAB-491 8 NRC 245 (1978)

and presented, of the state of the staff's program for resolving USI A-17, and more specifically on my understanding of such parameters as scope, schedule, priority, and resources allocated to that program. These parameters determine the rate of progress and actual results that can be achieved, or be reasonably expected, at any given time; they are, therefore, important indicators or measures of the adequacy of any USI program, and of the prospects for timely resolution of the issue involved.

Despite unfavorable developments that had occurred with respect to these important parameters in the systems interaction program in the months preceding the presentation of staff's testimony on Contention 7B in the Shoreham hearing, I had remained hopeful at that point regarding the ultimate outcome of events in the systems interaction area and regarding the prospects for resolution of USI A-17 on some reasonable and still acceptable schedule. But there has been further decline in the months since; and the cumulative effect is now such that I can no longer continue, in good conscience, to support the position that the staff's systems interaction program provides currently an adequate basis for the "justification for operation" conclusion required under North Anna, as indicated in my earlier testimony.

3. <u>Background and Baseline At Outset of the Program for Resolution of USI A-17</u>

As alluded to in the preceding, it is necessary to go back in time further than my participation in the <u>Shoreham</u> hearing last summer to set the background and to establish the baseline against which are drawn my current judgments regarding the adequacy of staff's systems interaction program. To recount briefly the relevant background, the judgment by staff management and the Commission that the systems interaction issue is a legitimate safety concern, serious enough to warrant designation as an Unresolved Safety Concern (i.e, USI A-17), was documented as early as 1977;[2] and a program for resolution of this issue was initiated in May 1978.[3] That initial judgment and action by NRC management in this regard was reconfirmed and reinforced in the aftermath of the TMI-2 accident by a strong recommendation of the Lessons Learned Task Force[4] (of which I was a member), and by further action by staff management and the Commission,[5] to strengthen the existing, on-going USI A-17 program. In early 1980, the Commission approved for inclusion in the TMI-2 Action Plan a provision for an augmented and expedited systems interaction program; and a separate, dedicated organizational unit (the Systems Interaction Branch) was set up within the Division of Systems Interaction, NRR to plan and coordinate the conduct of the new, augmented program. By mid-1980, the new Systems Interaction Branch had developed the

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[2] See NUREG-0410

[3] See NUREG-0510 at p. A-12

[4] See NUREG-0585, Section 3.2 and Recommendation 9

[5] See NUREG-0660, Item II.C.3

program plan for the augmented (II.C.3) systems interaction program.[6] The expanded program included (i) studies in which staff-developed methodologies were to be applied on a trial basis in selected plants late in the construction and OL licensing process, and (ii) other studies, (already committed to by the owners of the Diablo Canyon 1 & 2, and Indian Point-3 facilities, to be initiated in mid-1980 and early-1981, respectively) employing methodologies developed by the utilities involved. The results of all these efforts, taken together, were intended (i) to provide the basis for resolution of USI A-17, and for the development by the staff of additional requirements and regulatory guidance for systems interaction studies (if required) for application to all reactors, within about 2½ years, and (ii) to provide useful information and insights to be factored into decisions regarding implementation of the National Reliability Evaluation Program (NREP).[7]

With the preceding background (by way of further establishing the "baseline" alluded to earlier for current judgments of program adequacy) the decisions and actions taken by staff management and the Commission to this point in the systems interaction chronology can be characterized as follows:

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[6] See Memo, dated 11/21/80, Stolz to Rubenstein, "SIB/DSI FY 81 Resource Projection"

[7] See NUREG-0660, Item II.C.2

a. Baseline Consideration #1

The decisions and actions taken established the systems interaction program, in a very real sense, as a necessary regulatory activity i.e., as a USI program[8] which under existing rules must be addressed in reactor licensing safety evaluations.... (as contrasted to other highly desirable programs and activities, such as probabilistic risk assessment, safety goal development, etc., also provided for in the TMI-2 Action Plan, but which need not be so addressed)

b. Baseline Consideration #2

The decisions and actions taken indicated clearly that staff management and the Commission intended timely resolution of this important issue. The period of time in which it was thought initially that this could be accomplished was 1-1½ years. However, it was found that the fault tree methodology which had been developed in the pre-TMI phase of the USI A-17 program was not suitable for general, broader application in systems interaction analysis, (as had been counted on)[9]; so about a year was added to the time period that had initially been contemplated for program performance, to allow for search-and-development of possible alternative methodologies by the staff. It should be said, however, that allocation of even ~2½ years for resolution of such a complex unresolved safety issue necessarily implied and, indeed, required

[8] See NUREG 0510, at p. 10, p. 11, and p. 49 (Table 1-Category A definition)

[9] See Memo, dated 5/20/80, Angelo to Kniel, "Summary of Meeting with Sandia...to discuss...Task A-17"

assignment of high priority, and strong commitment to the USI A-17 program by staff management and the Commission.

c. Baseline Consideraton #3

With regard to the question implicit in the specification (as in Baseline Consideration #2, above) of the period of time to be allowed (at the outset) for the program to achieve timely resolution of USI A-17 (i.e., How to determine what is reasonable in that regard in view of the urgency of the matter?), the general concern underlying can be stated as follows:

> "Things unanalyzed" in the design of reactor plant systems (e.g., common mode/common cause mechanisms, and the effects of non-safety component failure) can lead to "things unexpected" in the operation of reactor facilities (e.g., occurrence of unanticipated events, including some serious enough to be termed accident precursors). And no matter how well trained or capable reactor operating personnel are (i.e., given some finite unreliability rate in operator actions), if the "unexpected" happens often enough (and it does, based on operating experience reports) for long enough, the likelihood of a serious accident (like TMI-2) can become unacceptably high.

The judgment, then, regarding what is a "reasonable" period of time to allow for resolution of tbe systems interaction issue involves somehow qualitatively (i) consideration of the rate of occurrence of unexpected events (in particular, serious precursor events) and (ii) a sense that the time allowed for resolving underlying causes of such events ought not to exceed some prudent fraction of the "average interval" for occurrence of such events, based on experience and observation. To say the obvious, that is a very difficult judgment for any individual to make, and should not, therefore, be left to ad hoc individual judgment. Such a difficult judgment on such a complex, important safety issue should properly be evolved (as <u>was</u> done in the series of events leading up to initiation of the II.C.3 systems interaction program; see Baseline Consideration #5) through a broad-based consensus forming process. As a strong corollary, once established in the proper manner (as described above, and in Baseline Consideration #5), schedules specified for the resolution of important safety issues (e.g., USI A-17) ought to be regarded seriously, and ought <u>not</u> to be overturned or extended significantly except on the basis of an equivalent process. More specifically, significant extensions should not be permitted or condoned simply by virtue of <u>default</u> on performance of the schedule established by consensus.

d. Baseline Consideration #4

Consistent with the high priority assignment and timely resolution objective for the augmented, post-TMI systems interaction program (see Baseline Consideration #2 above), although the II.C.3 program was to be closely coordinated with other programs (such as IREP[10] and NREP[11]), the schedules for the completion of studies intended to lead to the resolution of USI A-17 were established initially so as not to be linked to, or dependent upon, IREP/NREP program schedules in anyway that would delay achievement of the necessary USI-related objectives. Further indication of such intent is seen in the fact that the management of the systems interaction program (II.C.3) was established initially separate from the management of the IREP (II.C.1) and NREP (II.C.2) programs (i.e., with the program management involved in each case reporting to the Office Director and Executive Director levels through different chains of command).

e. Baseline Consideration #5

The decisions and actions taken in establishing both the initial USI A-17 program in 1978, and the augmented, post-TMI systems interaction program (II.C.3) in 1980, were taken within the context of an existing, established regulatory structure and process in which well-established (approved) deterministic criteria and requirements define what is adequate safety unless/until changed by due process

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[10] Interim Reliability Evaluation Program (IREP). See NUREG-0660, Item II.C.1

[11] See NUREG-0660, Item II.C.2

(i.e., the process outlined here). Those decisions and actions were based broadly on widely-shared qualitative judgments regarding the importance of the issue involved and the necessity for prompt action and timely resolution (see Baseline Consideration #3). The decisions involved were evolved through a highly-visible and open consensus forming process, which included full opportunity for review internally by cognizant NRC staff and ACRS.

4. Changes in Material Facts or Circumstances Affecting Testimony

Having established in the preceding the background and baseline which form the basis for my understanding of the staff's system interaction program, and against which I form judgments regarding its "status" and adequacy of any given point, I identify, in the following, significant changes that have occurred with respect to these baseline facts and circumstances which affect my earlier testimony. Some of the changes identified occurred before my Shoreham testimony, and some after; but all bear materially on the question of current validity of my earlier testimony. And I believe that all must be considered together to understand fully my current position in this matter.

a. Excessive Delay in Resolution of USI A-17

The most significant deficiency of the current system interaction program impacting the validity of my earlier testimony is that, although we are now nearly at the end of the period of time allocated for the resolution of USI A-17, we are nowhere near to achieving resolution of this important safety issue, along the current track and at the current pace. My optimistic estimate, in that regard, is that that goal is still 2-3 years off without significant reordering of priorities and re-constitution of the II.C.3 program along the lines suggested herein. I conclude, therefore, that the program cannot be regarded or characterized as adequate (specifically in the sense required to be addressed under North Anna; see Baseline Considerations #2 and # 3).

To be somewhat more specific, although notable progress has been achieved in the development of promising "candidate" systems interaction methodologies by the staff (as planned), demonstration or trial of those methodologies has not yet been done (or even begun). And while there have been hopeful developments recently with regard to getting those efforts underway finally (on the basis of initiatives taken/supported by the Director, NRR himself), it is clear that the completion of the demonstration phase of the II.C.3

program will take significantly longer to complete than initially planned (e.g., perhaps an additional 1-2 years). Also, although extensive, broad-scope systems interaction search efforts have now been completed at the Diablo Canyon and Indian Point-3 facilities using utility-developed methods, it now appears certain (i) that the planned submittal of unevaluated Indian Point-3 search results to the staff in late 1982 or early 1983, will now be delayed until late 1983 (due to hearing related considerations and complications), and (ii) that the final submittal of evaluated Diablo Canyon search results, which had been expected in late 1982 is now delayed indefinitely (due to well-known licensing-related difficulties that have arisen in that case).

In full view of these circumstances, the prevailing staff view seems to be to "stay the course"; i.e., continue along the current track at whatever pace can be achieved to eventual resolution of USI A-17, whenever that may occur. Under this view the program could be considered adequate currently simply because there is some systems interaction work currently underway (albeit well behind schedule), and because there is "no evidence" that drastic measures must be taken to hasten resolution of the system interaction problem. My view, instead, is that there is "no evidence" that the consensus judgments, regarding the seriousness of the safety concern involved and the need for timely resolution (i.e., in the time period allocated and agreed upon at the outset; see Baseline Considerations #2 and #3), were that wrong in the first instance. The decision to delay or extend the schedule for resolution of USI A-17 is, by its very nature, a major safety decision and should not be made by default, or by a few individuals on the ad hoc "no evidence" basis indicated. (See Baseline Consideration #3)

I believe, therefore, that the proper course of action at this point is (i) to recognize the inadequacy of the current state of the program, and (ii) to "call the question" for reconsideration, and submit it to the same decision making process that established initially the time to be allowed for resolution of USI A-17 (See Baseline Consideration #5). In that respect, I would favor strongly this time around a currently-appropriate variation on the original recommendation made by the Lessons Learned Task Force in 1980 in this regard,[12] and the similar recommendation made by ACRS in January 1982[13], to wit: Require all licensees and OL applicants to begin limited systems interaction reviews of their facilities immediately, using methods now known and documented for use or

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[12] See NUREG-0585, Section 3.2 and Recommendation 9

[13] See ACRS letter dated 1/8/82 ~~1/18/82~~, "Systems Interactions"; also see ACRS letter dated 3/9/82, "Report on SI Study for Indian Point -3."

trial (even though not completely evaluated at this time). The reasons for favoring now the more direct and immediate approach are (i) failure to resolve the systems interaction issue in the three years that have passed since inception of II.C.3 (or in the five years since USI A-17 was initiated) by employing a less direct and immediate approach, and (ii) clear indication now that licensees do not need to wait on the staff any longer to develop and demonstrate workable systems interaction methodologies that can produce safety-beneficial findings and results.

In this regard it is noted that, while the staff (for whatever the reasons) has not developed and applied workable systems interaction methodologies in the time allotted initially under the II.C.3 program, three utilities have done so (i.e., at Diablo Canyon, Indian Point-3, and most recently the Perry facility). Although the results of these efforts have not yet been fully-evaluated by the utilities involved and reviewed by the staff, in several instances on the basis of licensees' own prudent judgment, modifications to facility designs have already resulted from these system interaction reviews.

So a broad scale effort involving limited-scope systems interaction reviews in all operating facilities and NTOL plants could both (i) produce safety beneficial plant specific findings (as has already been done) and (ii) at the same time provide much more expeditiously and extensively actual systems interaction data and information needed by the staff for making final decisions regarding the possible need for more comprehensive systems interaction reviews generically. Suitable arrangements could be made between the staff and each utility regarding the scope of review to be done at each facility, and regarding the choice of methodology to be applied, (including choice of one of the staff's candidate methodologies, if mutually agreed).

As a final point regarding this particular aspect of changes in circumstances that have affected my earlier testimony, it might seem that the conclusions drawn at this time in this affidavit, regarding inadequacy of the program because of failure to resolve USI A-17 on the schedule initially established (i.e., about now), could have been drawn as easily 6-8 months ago as now (i.e., during the preparation and presentation of my earlier Shoreham testimony).[14] Such is not the case. Although (as alluded to in Section 2 above)

[14] See, for example, Transcript of TMI-1 Appeal Board proceeding at p.300, for for reaction of Appeal Board just to the changes of circumstance outlined for them in the affidavit cited in footnote 19.

there had been unfavorable developments in some aspects of the systems interaction program in the months preceding my participation in the hearing (described in further detail in Section 4.b following), the program in other important aspects was showing significant progress and results. For example (i) the Indian Point-3 systems interaction program plan was approved in early March 1982, and was underway and proceeding very well by early April, (ii) the matrix-based dependency analysis methodology development effort was launched in late Spring 1982, and (iii) prospects were very bright for the staff receiving extensive <u>actual</u> systems interaction review results from both Diablo Canyon and Indian Point-3 by late 1982. Additionally, there seemed to be real hope of getting the badly-lagging methodology demonstration phase of the program back on track and moving as a result of a development that occurred in early May 1982. At that time, there came down from the Chairman's office a request for a briefing on the status of the system interaction program. I interpreted this as a hopeful sign because it indicated a show of interest, <u>initiating</u> at the Commission level, in the state of the program; and it seemed a very real possibility that this timely show of interest from that level could result in a turning point, especially for the methodology demonstration program which was lagging at that point.

So it can be seen, I believe, that at the time of my involvement and participation in the Shoreham hearing there were still a number of reasons to support the (hopeful) view that the staff's system interaction program, although experiencing some serious difficulty, was still adequate at that point.

b. De-emphasis on Systems Interaction Program Objective

In March 1981, the Systems Interaction Branch (SIB) of the Division of Safety Integration (DSI) was abolished, and all but two of the nine SIB professionals working on systems interaction were assigned to other licensing-related activities within NRR. I was one of the two remaining former SIB members who were transferred to the Reliability and Risk Assessment Branch (RRAB) of the Division fo Safety Technology (DST) to try to continue the II.C.3 systems interaction program. RRAB is the organizational unit within NRR with lead responsibility for PRA-related activities, such as NREP.

The most obvious thing that can be said regarding this development is that, insofar as organizational "stature" and allocation of resources reflect the real importance ascribed and priority assigned to a given project/activity in the minds of NRC management, this development indicated a significant decrease in the perceived importance of systems interaction issue on their part, and correspondingly in the "effective" priority assigned to the program for resolving that issue. Concerns along these lines were expressed by me and other systems interaction staff to both SIB/DSI and RRAB/DST management at the time. And it was apparently also in this same vein that the TMI-1 Hearing Board raised questions regarding the motivation for, and possible effects of, this action.[15] All were reassured that any concerns in this regard were misplaced.

Despite such reassurances and the assumed good intentions underlying them, the effects of that action ultimately proved detrimental, as feared. Beginning at that point (gradually at first, but more noticeably as months passed) there began to develop in the management of the systems interaction program at all levels within NRR a noticable lack of emphasis on the completion of the II.C.3 systems interaction program (and resolution of A-17) on the basis and schedule established at the outset of that program.

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[15] TMI-1 Hearing Transcript at 15,615-15.629

More and more with time, the new organization seemed to lose sight of the fact that both the need and schedule for timely resolution of USI A-17 had been established at the outset by a broad consensus, based on the widely-shared judgment that the seriousness of the safety concern involved warranted an expeditious effort to resolve it. By contrast, at the same time that this apparent decline of emphasis and sense of urgency was occurring with respect to the systems interaction concern, increased visible emphasis was placed by staff management, and even the Commission, on PRA-related programs and activities. (e.g., quantitative safey goal development). It is in this respect that it simply must be said, at this point, that what has resulted is an inappropriate imbalance with regard to the importance being placed by RRAB/DST and NRR management currently on what is essentially "nice" (i.e., PRA-related activities) as compared to what must still be regarded, under existing rules and established procedures for reactor licensing, as "necessary" (i.e., programs for resolution of USI A-17).

These changes in attitudes on the part of management towards the importance, urgency, and priority of the system interaction concern are a major factor in my judgment of the adequacy of the systems interaction program currently, particularly with respect to prospects for resolution of USI A-17 at any reasonable time in the future, without a significant reordering of priorities and program redirection. (See Baseline Considerations #1, #2, #3, and #5).

The following specific examples are illustrative of the preceding general observations, I believe:

(1) Withholding/Delay of Final Approval for Implementation of Systems Interaction Methodology Demonstration

In October 1981, approval was given by DST to a proposal for initiation of the methodology demonstration phase of the II.C.3 program. In this proposal, approval by NRR was requested regarding final selection of the NTOL pilot plants in which candidate systems interaction methodologies were to be tested.[16] No action was taken (either approval or denial) by NRR at that time; and the effort stalled at that point, apparently over concerns that developed in connection with cost-benefit estimates required for the expected review by the Committee for the

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[16] See Memo, 10/28/81, Murley to Denton, "Implementation of Systems Interaction Interim Guidance".

Review of Generic Requirements (CRGR) of any NRR approval action on this proposal. In February 1982, however, in a letter from Mr. Dircks to ACRS (which required concurrence by NRR)[17] it was noted that "...the staff proposes to begin soon with reviews of four NTOL plants using two methodologies ..." That seemed surely to indicate some movement toward final approval of the proposal to initiate the studies described to the ACRS. However, more weeks passed with no final action on the request.

Meanwhile, (as also noted in the letter to ACRS), RRAB and DST management began considering various options for combining the systems interaction program with an already envisioned NREP/SEP combined review program. At this point still, the emphasis was said to be on expediting the resolution of USI A-17, as well as achieving cost-benefit advantages (to help in gaining acceptance/approval from (CRGR), by combining unnecessarily duplicative aspects of the three programs

[17]See Letter dated ~~2/21/82~~ 2/12/82, Dircks to Shewmon, "Systems Interactions".

done separately). Apparently the promise seen by NRR in this approach was great enough that NRR approval of the October 1981 DST proposal on initiaiton of the NTOL pilot plant methodology effort was delayed again, while the combined program idea was developed and explored further. That process has continued since;[18] but to date no final approval has been given by NRR for implementation of any methodology demonstration studies under any option. In the process, however, the initially proposed NTOL pilot plant alternative, approved by DST in October 1981 was discarded altogether. (I first learned that this was official in August 1982; a statement in this regard was inserted into an affidavit that I was preparing to the TMI-1 Appeal Board[19] in response to their request for a report on the status of the II.C.3 System interaction programs). As a final comment, it is noted pointedly that the notion of expediting the resolution of USI A-17 and achieving cost-benefit advantages by combining the program for resolution of USI A-17 with planned PRA-related programs did not work out well in any respect. I believe the basic error involved was in RRAB, DST and NRR management (i) not taking a more

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[18]See, for example, Memo dated 9/16/82, Ernst to Miraglia, "Revised CRGR Letter SEP Phase III/NREP", and Enclosures 1 & 2.

[19]See Affidavit dated 8/6/82, James H. Conran to TMI-1 Appeal Board.

aggressive posture with CRGR in presenting the II.C.3 related program proposal on its own merits, i.e., as a necessary program for timely resolution of a USI, and (ii) not resisting the post-facto imposition of a cost-benefit criterion in a way that delayed excessively the progress of that necessary program. (See Baseline Considerations #1, #2, #3, #4, and #5).

(2) Systems Interaction Analysis "Just a Part of PRA"
Even before being transferred to RRAB, I had begun to explore, in the context of my review of the Program Plan for the Indian Point-3 Systems Interaction Study the so-called systems interaction/PRA "interface", to try to understand better the relationship between the PRA which was already being performed (during 1980 - 1981) at the Indian Point facility and the proposed systems interaction study proposed at Indian Point-3.[20] As a result of my study of the interface question, I concluded, that the inter-system dependency information developed in a systems interaction analysis is important

[20]See Shoreham Hearing Transcript, at p. 7534.

in assuring the accuracy of PRA results; to such degree, in fact, that systems interaction analysis must be regarded logically as a prerequisite to PRA.[21] (ACRS also made a similar observation in January 1982).[22] In documenting my conclusions in this regard, and in discussing this matter with RRAB and DST management, however, I took great pains to point out even more importantly that systems interaction analysis has inherent value completely aside and apart from PRA; because its results can be used readily and effectively to improve safety (in the context of the current "deterministic" licensing approach), even if PRA is never done.

I objected explicitly to the tendency that I saw within the organization to think of system interaction analysis as "just a part of PRA," because that tends to subordinate systems interaction analysis (a "necessary" program under existing rules and established procedures for reactor licensing, for resolution of USI A-17) to PRA-related programs and objectives (which do not have

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[21]See "Meeting Summary and Status Report" for July 24, 1981 ..." by J. H. Conran, at p. 3-4.

[22]ACRS Letter, dated 1/8/82, "Systems Interaction"

that "necessary" aspect to them in the established system). The culmination of this tendency manifested itself, I believe, in the abortive efforts (described in 4.b (i) above) to combine the II.C.3 systems interaction program methodology demonstration studies with NREP, without regard to the impact on the schedule for timely resolution of USI A-17. (See Baseline Considerations #1, #2, and #4)

(3) Use of Unreviewed Risk-Based Decision Criterion

Another manifestation of the "way of thinking" addressed in 4.b(2) above, is the informal, ad hoc use of an unreviewed risk-based decision criterion in deciding important aspects of the USI A-17 program performance. It appears that this practice figured, at least partly, in the decision to withhold final approval on implementation of the methodology demonstration phase of the II.C.3 program. A partial basis cited recently for withholding final approval in that instance was that the systems interaction staff had not shown that the "risk benefit" to be gained by doing systems interaction analyses would be significant enough to justify the effort and expense of trying. Such reasoning amounts to overturning, without due process, a major safety decision made previously, on the basis of widely-share qualitative judgments, by post-facto application of an unestablished, quantitative risk-based criterion, (See Baseline Consideration #5). It is questionable also on the basis of the following considerations:

- Inadequate treatment of common-cause failure is an acknowledged major source of uncertainty in quantitative estimates of risk based on current probabilistic risk analysis methods.
- Systems interaction study is to a very great extent the pursuit of efficient methods to treat comprehensively and effectively common-cause or dependent failure.
- The use, therefore, of quantitative risk estimates based (necessarily) on current risk analysis methods (flawed as they are by uncertainties arising from inadequate treatment of common-cause or dependent failure), as a basis for deciding to delay or halt system interaction studies that could eliminate or reduce significantly such uncertainties, seems at, best self-defeating, and at worst questionable logically.

Said another way, USI A-17 must be resolved before either (i) the current deterministic licensing basis and process, or (ii) PRA and quantitative safety goals, can be applied with the improved confidence sought in reactor licensing today (because they are both "flawed" by the same source of uncertainty, i.e. common-cause or dependent failure. So we should get on with it. What we need now as before is an adequate program to address this "joint" problem expeditiously and effectively.

c. Shoreham Specific Considerations

It should be said that any concern regarding the adequacy of the staff's generic systems interaction program has added significance in the Shoreham case. It must be recalled that LILCO has taken the position that the PRA that has been performed at the Shoreham facility has, in effect, resolved USI A-17. It seems fair to conclude, therefore, that if the staff does not effectively pursue timely resolution of USI A-17 through its II.C.3 systems interaction program, the concern involved is not likely to be pursued further by positive dedicated programs by LILCO.

There is, further, another possible synergistic-type consideration arising from LILCO's position on the safety classification and safety classification terminology matter at issue between staff and LILCO (addressed in following sections of this affidavit). It is now clear that LILCO truly does not understand what is required minimally for safety, in the same way the staff (and the regulations) construe that phase. LILCO's position in that matter makes it less clear, then, whether systems interactions concerns have been treated adequately at Shoreham. For example, it may be that the difference between the positions of LILCO and the staff, regarding the claim that the Shoreham PRA resolves satisfactorily (for Shoreham) the systems interaction concern, derives from this fundamental difference in understanding of what is required minimally for safety (i.e., "How little, actually, is enough?") rather than from theoretical, matters-of-degree type arguments regarding the question "How far beyond what-is-required is enough?" (as seemed to be suggested in the discussions at the hearing regarding dependency analysis and walkdowns in the Shoreham PRA)[23] This question would seem to bear heavily on the determination of whether LILCO has satisfied what is required under North Anna, regarding USI A-17, especially in this situation where the staff's "contribution" in that regard is called into question.

[23]See Shoreham hearing transcript at p.6653, p.7500, p.7634 and p.7847

## SAFETY CLASSIFICATION TOPIC

6. General Statement of Amendment to Testimony

At the time of my participation in the Shoreham hearing, it was not clear to me, as it is now,( with more time to consider thoroughly all of the testimony of Applicant's witnesses, and its full implications) that LILCO truly does not understand what is required minimally for safety by NRC under the regulations (i.e., what is considered necessary and sufficient to provide reasonable assurance of no undue risk to the health and safety of the public in the operation of a facility). Coming to the discussions of these matters in the hearing with the background described extensively in my testimony, I was predisposed to think of the defect in Applicant's stated position regarding the safety classification term "Important to Safety" as simply a "language problem". That is to say, at bottom, I believed that, although we subscribed to a different set of words to describe them, both the staff and Applicant understood in basically the same way the fundamental safety concepts underlying the terms "Important to Safety" and "Safety-Related" (as the staff apply those terms). Considerable effort was made by counsels for the staff and Applicant, while Contention 7B was being argued, to work out what were perceived as resolvable language differences (as contrasted to fundmental lack of mutual understanding regarding what is required minimally for safety). I participated in those efforts, and upon several occasions responded to cross-examination by counsel for Applicant in that context and spirit, suggesting that we may have achieved near-meeting of the minds by the end of argument of Contention 7B. I recognize now, that we are, in fact, <u>not</u> near a meeting of the minds on the very important fundamental safety concept at root in this matter. As a general statement of amendment, therefore regarding my testimony in that respect, it should be said that, to the extent that the Board or Parties might rely on such statements regarding "meeting of the minds" in <u>my</u> hearing testimony to determine outcome on Contention 7B, they should <u>not</u> do so.

7. <u>Basis for Amendment of Testimony</u>

The further understanding that I have developed in this regard is based on the following:

a. opportunity to consider longer and review more thoroughly the testimony of Applicant's witnesses,

b. involvement in the review of recent proposals by LILCO to the staff for resolving differences left outstanding at the end of argument of the safety classification and safety classification terminology issue in the hearing, particularly regarding non-safety Q.A.

c. synergistic consideration of a) and b).

In that context I was struck by how little movement could be seen in LILCO's six month old differences with the staff on these matters.

With a license at stake, and that long to think about and work on it, it seemed remarkable to me that there would not have been more substantive effort on LILCO's part to develop or promote improved mutual understanding on what I had thought were only language differences. The staff, for example, has continued the effort to develop a listing of "Important to Safety" structures, systems and components; and, recently, a draft report containing preliminary results of that effort has become available.

In pondering these questions further, I carefully reviewed the testimony of Applicant's witnesses again (in particular, testimony at p. 5425-5449 of the <u>Shoreham</u> hearing transcript), in which staff counsel sought to establish by cross-examination equivalency between staff's and Applicant's understanding of the fundamental safety-concepts involved, even though the language applied was different. In that review, I finally recognized that, in responding to counsel's questions, Applicant's witnesses invariably couched their responses in a way that acknowledged <u>some</u> safety relevance to the specific examples provided by counsel of things "Important to Safety, but not Safety-Related", but carefully avoided acknowledgement or recognition that such items had <u>enough</u> safety relevance or importance to number them among that category of things required <u>minimally</u> for safety by the regulations.

8. Implications of Amendment to Testimony

Having come to this realization and fuller understanding of these matters, I believe the full implications of this can be summarized as follows:

a. The concerns that occupied me chiefly at the time of the hearing focused most heavily on the implications of language differences, (i) with respect to impact on staff's ability to rely on Applicant's affidavits in the audit review context, thus complicating significantly (if not prohibitively)staff's ability to come to a finding of "reasonable assurance..." through the usual, established audit review process, and, (ii) with respect to possible impact on staff's ability to obtain information required for its regulatory function during operation of Shoreham, as contemplated under Part 21 (because the Applicant might not realize that he had to report information regarding failure of some component which he did not "call" Important to Safety, but staff did).

b. My concern at this point is more serious, however. I no longer believe that our differences involve only a language problem to be sorted out mechanically. There now appears to be a substantive defect in Applicant"s true understanding of what is really required minimally to protect public health and safety. A language problem could be remedied simply by imposition of a definition; (or possibly even by a much more complicated alternative scheme proposed by LILCO). But understanding of the fundamental safety concepts underlying the usage of the term "Important to Safety" in the regulations cannot be imposed, (as for example by a condition to license). Understanding must be developed, and demonstrated, I believe.

Therefore, I believe that a condition for (i.e., prerequisite to) a license in this case should be development by LILCO of a listing of "Important to Safety" structures, systems and components for Shoreham, as a vehicle and means for developing and demonstrating the requisite understanding of what is required minimally for safety in the operation of Shoreham. In the construction and design phase, the very detailed SRP and Regulatory Guide information can perhaps provide a "safety net" or "backstop", to mitigate serious misunderstandings regarding staff's (and the regulations') safety classification terms. However, in the operation of a facility there is little that would act effectively in a similar way (i.e., as a backstop), either in the regulations, or in staff's procedures and activities. There must be understanding of what is necessary minimally for safety as a prerequisite for safe operation. And because Applicant's understanding in that regard is so clearly called into question, by their own testimony, I believe there should be demonstration of remedy before licensing. The staff's preliminary (draft) listing of structures, system and components "Important to Safety" (referred to above) could be used as the starting point of an effort to do that, and could enable completion of such effort on a basis that would not have to interfere with licensing schedule.

C. Need for Additional Testimony

Mr. Conran's February 9, 1983 affidavit, if received in evidence, will significantly modify evidence proffered by the Staff in support of its position in the proceeding. Fairness requires that the Staff be permitted the opportunity to supplement the record directly affected by Mr. Conran's modification of his position. The Staff is prepared to offer additional testimony on each of the two subjects addressed by Mr. Conran's affidavit. The receipt in evidence of this additional testimony is in the interest of a full and fair hearing record upon which a decision can be made. The Staff proposes to offer this testimony by affidavit.

1. Systems interaction (A-17)

The Staff is preparing additional testimony on the subjects of the status and progress of the Staff's program in support of unresolved safety issue A-17 and the basis for the Staff's position that Shoreham can be operated safely despite the pendency of unresolved safety issue A-17. That testimony will be sponsored by Ashok C. Thadani, Branch Chief of the Reliability and Risk Assessment Branch, and Franklin D. Coffman, Section Leader of the Systems Interaction Section within the Reliability and Risk Assessment Branch. The principal points of that testimony are expected to be as follows:

1. the Staff's current licensing requirements provide reasonable assurance of no undue risk to public health and safety from potential adverse systems interactions;

2. unresolved safety issue A-17 is confirmatory in nature;

3. the Staff's program on A-17 is progressing satisfactorily toward resolution;

4. no plant-specific systems interaction analyses are or should be required until completion of the Staff's program determines whether they are necessary and justified.

ENCLOSURE 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 0 9 1979

MEMORANDUM FOR: Commissioner Bradford

THRU: Roger H. Moore, Chief, Applied Statistics Branch, MPA RHM
Norman M. Haller, Director, MPA
Lee V. Gossick, Executive Director for Operations

FROM: David Rubinstein, Applied Statistics Branch, MPA

SUBJECT: PROBABILITIES THAT THE NEXT MAJOR ACCIDENT OCCURS WITHIN PRESCRIBED INTERVALS

Apparently your request of January 31, 1979, to Saul Levine and John Austin for "the correct way to state the statistical significance of ... 400 reactor years of operation without major accident" has received fairly widespread attention in the Commission. I believe most of the concerned persons addressed this problem in terms of upper confidence limits for the rate of a major accident or from the point of view of hypothesis testing. An alternative way of addressing this problem is through a prediction interval. As used here the prediction interval focuses on the next major accident. It puts an upper bound on the probability that the next major accident will occur within a specified number of reactor years. Alternatively, it will give a lower bound of the probability that the next major accident will occur after a specified number of reactor years. On the basis of some assumptions discussed below we may say for example:

a) The probability is less than .5 that the next (i.e., the first) major accident occurs within the next 400 reactor years.

b) The probability is less than .05 that the next major accident occurs in the next 21 reactor years.

c) The probability is larger than .5 that the next major accident will occur after the next 400 reactor years. This is equivalent to statement (a).

The column headed by I in Table 1 and the graph with the triangles in Figure 1 give more detailed results of the prediction interval method. The results are given for both reactor years and time expressed in calendar years; a calendar year is taken equivalent to 70 reactor years. At present there are approximately 70 commercial operating reactors.

8pp

ENCLOSURE 3

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The results suggest perhaps unwarranted pessimism because of

a) conservative features in the analysis

b) large statistical variability of times to first (or next) major accident

c) lack of engineering considerations.

The prediction intervals are derived under the assumptions that major accidents occur as a Poisson process; i.e., at a constant rate and independently. These assumptions provide perhaps a reasonable approximation; however, this is not readily demonstrated. The prediction interval as used here does have one conservative feature in that it "equates" the time to the first major accident with the time cumulated to date without major failure. My gut feeling is that this conservatism is likely to outweigh possible non-conservatisms in the assumptions. [I also see the possibility of obtaining more assurance based on somewhat plausible speculation about such matters as early high failure rates (often called infant mortalities) or relative occurrence rates of subclasses of accidents. Careful and detailed examination of existing failure data might provide support for such speculation.]

On the basis of the simplistic assumption of a Poisson process one can readily compute the probability distribution of the time to the next major accident for any specified value of the occurrence rate of major accidents. [In contrast the method of prediction intervals does not require knowledge or postulation of the value of occurrence rates to make probability or confidence statements about the waiting time to the first occurrence.] Table 1 and Figure 1 also provide probabilities relating to the waiting times to the next major accident for the major accident rates listed below:

A) One per 170 reactor years; this corresponds to the 90% upper confidence limit based on 400 reactor years with no major accident.

B) One per 580 reactor years; this corresponds to the 50% upper confidence limit based on 400 reactor years with no major accident.

C) One per 4,000 reactor years; this is an approximate amalgam of WASH-1400 and other "upper bounds", and the 50% upper confidence limit based on about 2,800 reactor years without major accident. Some persons make claims of from 2,500 to 3,000 relevant (in some sense) reactor years free from major accidents.

D) One per 1,000,000 reactor years; this has been included here as some sort of holy grail.

In line with your request, I took a very pragmatic approach and deliberately played down the theoretical aspects. I shall be happy to try to clarify them if you so desire; in particular, the precise interpretation of prediction intervals. I am appending a brief mathematical derivation in case mathematically inclined persons will read this memo.

One can duplicate similar computations for the time to the second major failure, third major failure, etc. To some persons the picture for later major accidents might appear somewhat less alarming. Again if you have interest in such computations, the Applied Statistics Branch can provide these.

David Rubinstein

David Rubinstein
Applied Statistics Branch
Division of Technical Support
Office of Management and
Program Analysis

cc: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Ahearne
Lee R. Abramson
Dan Lurie
Susan B. Young
Saul Levine
John Austin

## LEGEND TO TABLE 1

Probabilities that the Next Major Accident Occurs Within Prescribed Intervals

t is time expressed in either reactor years or calendar years.

$P(Y \leq t)$ is the probability that the next major accident occurs at or before time t.

Column I gives upper bounds for $P(Y \leq t)$ as obtained by the prediction interval method on the basis of 400 reactor years free from major accidents.

Columns A, B, C, and D give exact values for $P(Y \leq t)$ for given occurrence rates of major accidents as explained on pages 2 and 3.

$\theta$ is the reciprocal of the failure rates used in columns A, B, C, and D. It is also the mean time between major accidents.

## LEGEND TO FIGURE 1

Probabilities that the Next Major Accident Occurs Within Prescribed Intervals

The time scale in reactor years is given on the bottom and in calendar years on top. These scales are logarithmic and scientific notation is used for the large numbers; thus 5(4) is $5 \times 10^4$ = 50,000.

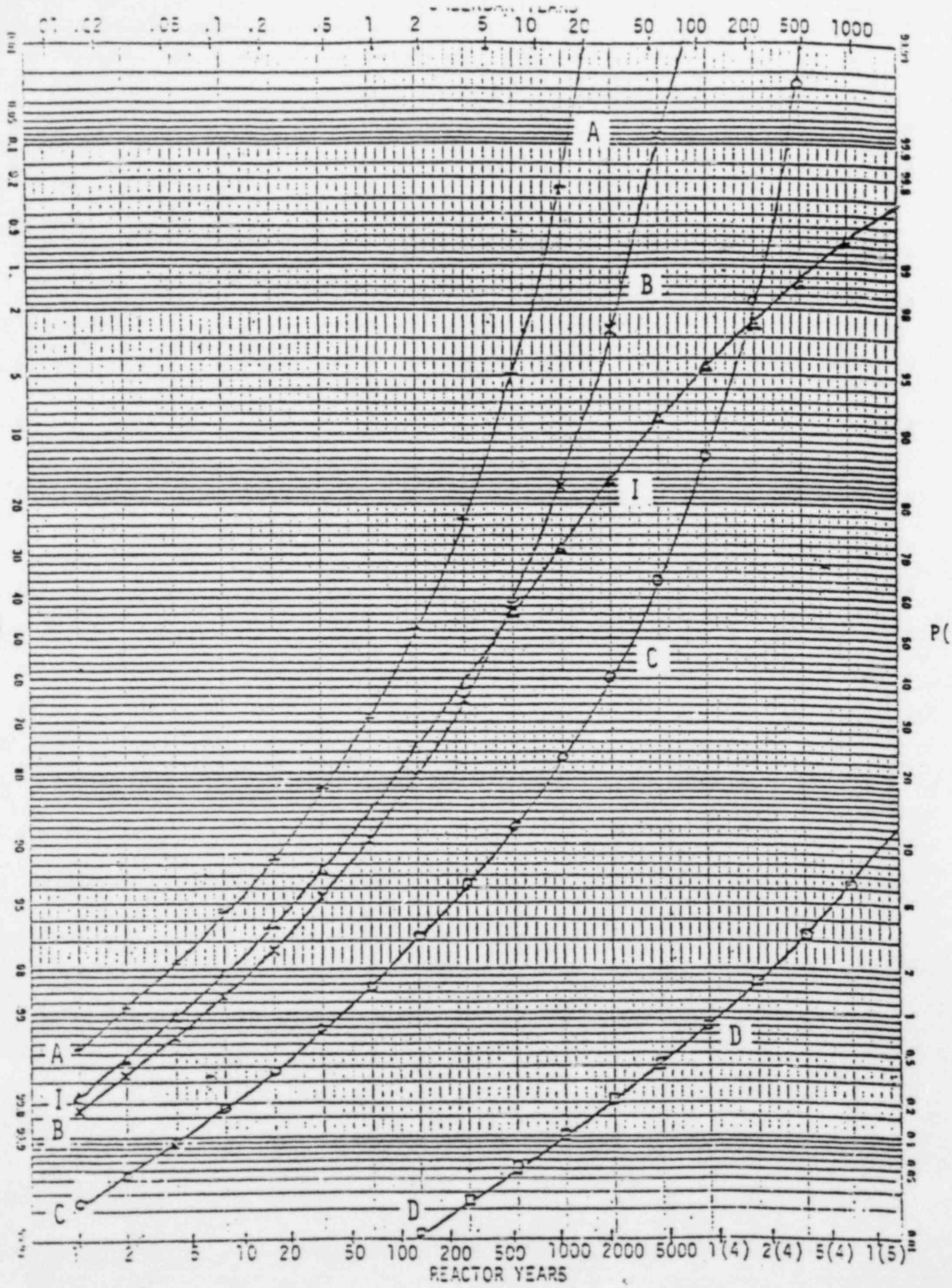
The graphs are plots of $P(Y \leq t)$ against the P scale on the right. The Q scale gives the complementary probability $1 - P = P(Y \geq t)$. Both scales express probabilities in percent; neither scale is linear.

The graph passing through the triangles ($\Delta$) refers to the prediction interval. The other graphs are: + for case A($\theta$ = 170); X for case B($\theta$ = 580); O for case C($\theta$ = 4,000); and □ for case D($\theta$ = 1,000,000).

TABLE 1

PROBABILITIES THAT THE NEXT MAJOR ACCIDENT OCCURS WITHIN PRESCRIBED INTERVALS

| t | t | $P(Y \leq t)$ | | | | |
|---|---|---|---|---|---|---|
| REACTOR YEARS | CALENDAR YEARS | I | A | B | C | D |
| | | | θ 170 | 580 | 4000 | 1000000 |
| 1. | .01 | 0.002 | 0.006 | 0.002 | 0.000 | 0.000 |
| 2. | .03 | 0.005 | 0.012 | 0.003 | 0.000 | 0.000 |
| 5. | .07 | 0.012 | 0.029 | 0.009 | 0.001 | 0.000 |
| 10. | .14 | 0.024 | 0.057 | 0.017 | 0.002 | 0.000 |
| 20. | .29 | 0.048 | 0.111 | 0.034 | 0.005 | 0.000 |
| 50. | .71 | 0.111 | 0.255 | 0.083 | 0.012 | 0.000 |
| 100. | 1.4 | 0.200 | 0.445 | 0.159 | 0.025 | 0.000 |
| 200. | 2.9 | 0.333 | 0.692 | 0.293 | 0.049 | 0.000 |
| 500. | 7.1 | 0.556 | 0.947 | 0.580 | 0.118 | 0.000 |
| 1000. | 14. | 0.714 | 0.997 | 0.823 | 0.221 | 0.001 |
| 2000. | 29. | 0.833 | 1.000 | 0.969 | 0.393 | 0.002 |
| 5000. | 71. | 0.926 | 1.000 | 1.000 | 0.713 | 0.005 |
| 10000. | 140. | 0.962 | 1.000 | 1.000 | 0.918 | 0.010 |
| 20000. | 280. | 0.980 | 1.000 | 1.000 | 0.993 | 0.020 |
| 50000. | 710. | 0.992 | 1.000 | 1.000 | 1.000 | 0.049 |
| 100000. | 1400. | 0.996 | 1.000 | 1.000 | 1.000 | 0.095 |
| 200000. | 2800. | 0.998 | 1.000 | 1.000 | 1.000 | 0.181 |
| 500000. | 7100. | 0.999 | 1.000 | 1.000 | 1.000 | 0.393 |
| 1000000. | 14000. | 1.000 | 1.000 | 1.000 | 1.000 | 0.632 |
| 2000000. | 28000. | 1.000 | 1.000 | 1.000 | 1.000 | 0.865 |
| 5000000. | 71000. | 1.000 | 1.000 | 1.000 | 1.000 | 0.993 |


CALENDAR YEARS
.01 .02 .05 .1 .2 .5 1 2 5 10 20 50 100 200 500 1000
A
B
I
C
D
P(
REACTOR YEARS
1 2 5 10 20 50 100 200 500 1000 2000 5000 1(4) 2(4) 5(4) 1(5)

Mathematical Derivation

The prediction interval based on accident free observation time resembles Laplace's Law of Succession. However, the following derivation is independent of the Law of Succession.

Assumption: Major accidents occur as Poisson sequence with parameter $\lambda$.

Definitions:

1) $t_o$ is an arbitrary exposure time to risk of major accidents.

2) $X_{(1)}$ is the time to the first major accident. Note that $X_{(1)}$ may be smaller or larger than $t_o$.

3) $X^* = \min[t_o, X_{(1)}]$ .

4) $Y = X(t_o)$ is the waiting time to the next major accident starting from $t_o$.

Theorem: For $k > 0$, $P[Y \leq kX^*] \leq 1 - \frac{1}{k+1}$ .

Proof: From the assumption of a Poisson process it follows that Y is independent of $X^*$. For positive k, the probability

$$P[Y \leq kX^*] = \int_0^\infty P[X^* \geq y/k]\,\lambda\exp(-\lambda y)dy$$

$$\leq \int_0^\infty P[U \geq y/k]\,\lambda\exp(-\lambda y)dy \text{ ,}$$

where U is a random variable exponentially distributed with parameter $\lambda$. Note that $X^* \leq X_{(1)}$. Also,

$$\int_0^\infty P[U \geq y/k]\,\lambda\exp(-\lambda y)dy = P[U \geq Y/k] = P[Y/U \leq k] \text{ .}$$

Since Y/U has an F distribution with 2 and 2 degrees of freedom and the density of $F_{2,2}$ is $f(t) = (1 + t)^{-2}$, it follows that

$$P[Y \leq kX^*] \leq 1 - \frac{1}{k+1} . \tag{1}$$

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 04 1981

MEMORANDUM FOR: Those on Attached List

FROM: David Rubinstein
Applied Statistics Branch
Office of Management and Program Analysis

SUBJECT: A STATISTICIAN'S VIEW OF NRC STATISTICS - A PRESENTATION TO THE ASA ADVISORY COMMITTEE ON NUCLEAR RESEARCH

The enclosed transcript of my talk to the ASA Ad Hoc Committee on Nuclear Research might be of interest to you.

David Rubinstein

David Rubinstein
Applied Statistics Branch
Office of Management and Program Analysis

Enclosure:
As stated

cc: See attached list

~~8307260540~~

ENCLOSURE 4

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NRC

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## A STATISTICIAN'S VIEW OF NRC STATISTICS*

MR. RUBINSTEIN: Since Carl started to talk about where I fit in, let me say that I belong to the Applied Statistics Branch which is the central consulting group on statistical problems, serving all of NRC. We are not part of the risk assessment group, and we have been relatively little involved in risk assessment. Sometimes we get involved either because we push our noses into it and occasionally because we are asked to.

As I was listening to the various speakers. I wanted to change my speech, but I sort of gave up. I may repeat certain things which other speakers have said. Please forgive me for that.

I should like to start on an upbeat note. There have been improvements in NRC uses of statistics. I seem to sense a refreshing quickening of pace; at this meeting and at a meeting last month on risk assessment, I noticed much more concern for the subleties of statistical problems and much more recognition and acknowledgment that the past performance has been less than perfect. This quickening of pace, might be called acceleration -- I should like to call it "a jerk." In common speech a jerk is a sudden change in force or acceleration. Engineers use the word "jerk" or "jerk function" to denote the derivative of acceleration. One might speculate that the current large value of the jerk function is caused by anticipation of the judgment which you are going to pass. Regardless of the merit of this speculation, the very fact that NRC has invited you to look over its shoulder is an extremely good omen. I am confident that your advice and guidance will help to keep the jerk, and perhaps the jerk of the jerk, positive for considerable time to come.

(Laughter)

A review of statistics at NRC is an ambitious undertaking I cannot do justice to. It covers a considerable time span; it covers many practitioners at NRC as well as practitioners outside of NRC. The latter include groups working under contract to NRC, as well as employees of vendors and licensees who are required to demonstrate some aspect of performance or safety. Often NRC licensing does rely on analyses performed by outsiders.

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*An edited transcript of a talk given by David Rubinstein on Nov. 7, 1980 in Washington, D.C., to the ASA (American Statistical Association) Ad Hoc Committee on Nuclear Regulatory Research.

Despite the introductory and sincere upbeat note, there is still much room for improvement in NRC's statistical applications. I shall deal in a broad brush fashion with some of the troublesome issues. I want to emphasize the word broad brush; NRC statistics is not a simple monolith. I indicated already evolution over time and that statistics is practiced in one form or another by many individuals within and outside NRC. Obviously the various individuals and groups do not perform uniformly well, nor does each perform uniformly well in all instances.

Despite diversity of application and quality, some deficiencies can be found rather frequently in statistical applications in the nuclear field. First I shall speculate on why there are rather frequent shortcomings, and then discuss some of the specific issues, and finally end with some more or less philosophical questions.

I believe that the penetration of the AEC and NRC by statisticians has been minimal and rather late. The full subtlety and complexity of statistical problems in the nuclear field has not been appreciated by many in the nuclear field, and this includes managerial personnel. There is widespread belief that physical scientists with some acquaintance of statistical methods can handle statistical problems adequately. This point relates to my next observation.

A technological, or perhaps even technocratic attitude, seems rather prevalent in the nuclear field. If there is a problem, there is a technological fix. Associated with this attitude or philosophy is an action-oriented approach that is not overly concerned with intellectual considerations; pragmatic considerations will do. Oftentimes, I am concerned whether the methods are even good pragmatism. The activist approach, whether pragmatically sound or not, is reinforced by pressure to provide answers and to provide them quickly.

Before I turn to some of the specific issues, I would like to point out that statistics is at least a moderately successful science because of somewhat precise concepts and somewhat rigorous methodology. Unfortunately, in the nuclear field, statistical concepts and methods get often blurred. It is now well known that the Lewis Committee called WASH 1400 inscrutable. Leaving WASH 1400 aside, I find much of the statistics in the nuclear field inscrutable, or vague. In fact, I -- and I believe other statisticians will -- find some analyses inscrutable, vague, questionable or wrong that might not have been regarded so by the Lewis Committee.

Yesterday, we already noted the confusion of rates, probabilities, and expected values. There were some incisive comments made about choosing distributions for maximum floods, and I would like to note that it was an NRC non-statistician who pointed out that we are frequently concerned with mixtures of populations.

Among other items of concern I find:

1. Confusion between random variables and parameters is common even when no Bayesian approach is intended.

2. Best estimate is a term that is extremely vague and frequently used. It could come from data or from subjective belief. It could be a mode, a mean, a median a 50 percent confidence limit -- usually an upper confidence limit -- or what strikes somebody as best without clear elucidation of what is best. It may only be a matter of coincidence that the best estimate is the minimum variance estimate in a particular class of estimates.

3. You will often hear the word "uncertainty"; a term that nuclear people seem to be particulary fond of. The first major technical report I reviewed in NRC used the word "uncertainty" where I think the following terms might have been more appropriate:

a) Random variable, or perhaps a variation thereof such as random error or measurement error;

b) Standard deviation;

c) Confidence limit;

d) Error or bias; and perhaps here one could even become more specific whether this was an error in a parameter value or an error in the mathematical model;

e) It was also used in that report for the density or distribution function.

At times I just did not know what the intended meaning of the word was, and I am not sure that the authors always knew what they were talking about. Other uses and misuses include the following:

a) The word "uncertainty" is also used as an equivalent to what is called an upper bound, and this term is not well-defined. It seems to denote a large or very large value in a not-clearly-specified set of values. In nuclear jargon, "upper bound" is not a literal upper bound.

b) Finally, "uncertainty" is used as sort of a catchall phrase for what one might call Bayesian uncertainties.

c) And, lo and behold, sometimes "uncertainty" is used as the condition of being in doubt. This particular usage of the word I prefer.

d) Regardless of the varied nature of the uncertainties, they often are sum-root-squared to yield "total uncertainty." While there is frequent recognition in the nuclear field of the diversity of uncertainties, much sloppiness and confusion still exist.

Now let me turn to some methodological problems. Bayesian statistics of one sort or another is widely used. The material that was distributed to you contains some examples and critiques. I do not wish to elaborate on these in detail. However, even at the risk of repetition, I should like to point out that often the Bayesian framework is not clearly formulated and there is considerable sliding between Bayesian statistics and frequentist statistics, and it can go both ways perhaps through several cycles in a particular analysis. Often there is no explicit mention or an indication of an a priori distribution, and even when a report starts with an a priori distribution, it may not end with an explicit posteriori distribution or probability. The probabilities seem to have become absolute.

Another technical problem of NRC is that of components or variance. Often we work in NRC with generic values which are evaluated and applied over presumably similar classes. Plant to plant variabilities may be ignored, or differences between different components may be ignored. The ignored variation may be the dominant contributor to variability and Bill Vesely in his talk clearly recognized this. There has been some progress in dealing with the complex random structure of the things NRC has worked with, but more systematic exploration, clarification, and proper analysis of the random structure is indicated. Model II, or mixed models of the analysis of variance are not well recognized in NRC. For that matter, Model I may be ignored.

As Bill Vesely pointed out, human factors and common cause problems are important, or perhaps even dominant contributors, to risk. It is extremely difficult to model these convincingly and to find appropriate data for estimation of parameters. In NRC terminology, there are great uncertainties with respect to these areas. This leads to some philosophical questions.

In view of large and not necessarily well understood uncertainties, can one properly quantify uncertainties? How should numerical analyses be used? How should they be communicated? The subject of communication is a large one in itself.

A related question is what should be the proper role of subjective judgment in governmental policy and regulation and how should one deal with subjective judgments and how should they be presented to the public and the political representatives?

I was going to stop here, but Mike Cullingford stimulated me to say that perhaps we shouldn't argue about whether we want Bayesian statistics or classical statistics. Perhaps we should ask the question, what is good scientific inference?

Thank you.

DRAFT

October 26, 1981

# Random Thoughts on Uncertainties, Risk Analysis, and Nuclear Regulation

David Rubinstein

Note: This paper reflects the views of the author. It should not be construed as a policy statement of the Nuclear Regulatory Commission.

ENCLOSURE 5

12 pp

DRAFT

# Random Thoughts on Uncertainties, Risk Analysis, and Nuclear Regulation

David Rubinstein

## 1. Introduction

Okrent's testimony [Ref 1] stimulated me to write some of my thoughts on uncertainty, risk analysis, and judgment in nuclear regulation. These thoughts represent my personal view. Except as noted below, I do not attempt to review other people's work in this area. I shall use Okrent's testimony and a paper [2] cited therein as a point of departure for developing my views.

I shall offer my reaction to the issues raised by Okrent and develop my own perspective on those issues which I think NRC should consider. This report has been written primarily for NRC insiders. However, to accommodate likely outside readers, I may explain some matters familiar to NRC personnel. Here I convey the topic of Okrent's testimony by citing its opening sentence.

> "My understanding is that the general focus of discussion for my appearance before the Nuclear Safety Oversight Committee today is to be the matter of how the Nuclear Regulatory Commission (NRC) makes decisions concerning the public health and safety in the presence of very considerable technical uncertainty."

Before getting on with the subject matter I should like to acknowledge that I have had little direct responsibility for probabilistic risk assessment at NRC. I have been an observer more or less on the fringes of NRC risk assessment. While this may indicate shortcomings, it may also provide a detached perspective.

In Section 2, I shall comment on Okrent's testimony and on how it fits into the evolution of probabilistic risk analysis at NRC. I discuss my personal view of the role of uncertainty in nuclear regulation in Section 3. Section 4 will elaborate on some shortcomings of analyses other than large uncertainties; it also deals briefly with the role of judgment in regulation and Okrent's call for criteria for judgment.

## 2. A Partial Review of Okrent's Testimony and of the Evolution of Risk Analysis at NRC

First I should state that I greatly appreciate Okrent's testimony. It comes closer to my own philosophy than any other NRC document I have read on this subject matter. On basic issues of NRC decision making, Okrent and I may be cousins, but not necessarily kissing cousins. I look upon Okrent's testimony as a stage in an evolution of how AEC/NRC deals with risk. I shall offer a grossly over-simplified outline of this evolution. This outline may do injustice to those who, in NRC jargon, have remained determinists or who have, contrary to the prevailing tendency (or just lip service), drifted toward "determinism". (In NRC, the "determinists" place relatively little reliance on probabilistic risk analysis for regulatory purposes; they rely primarily on "judgment".)
The evolution has the following stages:

A. "All-is-safe" stage: The conception that nuclear power is safe with relatively little formal analysis as backup.

B. "Wash-1400" stage: Development and frequent use of "scientific" or "technical" approach primarily by "probabilistic risk assessment" with loads and loads of fault trees. Still marked disagreement on the relative merits of judgment and probabilistic risk assessment.

C. "Post-Lewis-Committee" stage: Greater reliance on probabilistic risk assessment (and peer review) with strong admonition for evaluation and statement of uncertainties - even more fault trees.

I shall discuss below how Okrent is at least in an advanced part of the Post-Lewis-Committee stage.

In his introduction Okrent notes that there is profound uncertainty in many regulatory activities. In the body of his testimony he makes the additional points related to specific cases of NRC regulatory activity or the lack thereof. Among these are:

1) Advocacy for more plant specific analysis and decisions.
2) Tough questioning of action criteria related to probabilities of severe core damage, and of permitting plants to be operated under some stated circumstances.
3) Critique of imprecise terminology used in risk assessment.
4) Critique of inconsistencies in regulatory prescriptions.

Observations such as these have been made before. What gives Okrent's testimony special force is the cohesiveness and toughness of the entire testimony. It is in contrast to more typical (and lenient) attitudes and standards under the Post-Lewis-Committee stage. The final portion of Okrent's concluding comments also seem to indicate a change. To quote:

> "...Despite these <u>potentially serious difficulties</u> with probabilistic analysis, it <u>appears</u> that an <u>effort</u> to quantify the risks, or the increment in risk, associated with a particular safety issue is a worthwhile <u>part</u> of the process leading to decision. The assumptions must be clearly stated. The uncertainties should be defined, as <u>possible</u>. <u>Criteria for judgment should be developed and independent peer review should be used</u>.
>
> "Ah yes, how should we expect the NRC to make decisions on matters like hydrogen and non-seismically qualified auxiliary feedwater systems, which involve an atmosphere of technical uncertainty?
>
> "<u>With difficulty.</u>" [Emphasis added.]

The words I underlined seem to convey something less than an absolute faith in probabilistic risk assessment. First, I shall briefly discuss the reference to judgment. Even in the Post Lewis tradition, judgment has been called for with varying emphasis ranging from the notion that probabilistic analysis should categorically supplant judgment to actual reliance on probabilistic analysis only if it confirms one's prior judgment. Such extreme positions may be rarely stated, but I believe they come close to some persons' outlook or behavior. Undoubtedly, Okrent calls for judgment and he calls for the development of criteria for judgment. One plausible implication of that recommendation is that probabilistic risk assessment is not always trustworthy; therefore we must use judgment and attempt to rationalize and formalize the judgment process. The call for the development of criteria for judgment again seems to go beyond Post Lewis stage. I shall return later to this aspect.

I also wish to comment on the final phrase "With difficulty." I have heard speakers in the Post Lewis era give eloquent and penetrating description of the difficulties with probabilistic risk assessment and yet conclude with an optimistic prognosis of its use. What Okrent finds is difficulty from beginning to end. However, in the context of his testimony the phrase "With difficulty" seems to imply a hope or expectation of success. Because of this implication, I may part company with Okrent. In any case the phrase is not precise. Nor is the sentence "...to make decisions on...non-seismically qualified auxiliary feedwater systems." Is it just a matter making decisions or is a matter of making good decisions-with difficulty? (Presumably, Okrent meant good decisions.) What does difficulty mean? Can NRC solve the problem in one year, or in ten years. How many man-years or man-millenia are required? Would it require giant shake tables on which critical configurations of piping could be stressed with simulated earthquakes? Would it require eight full scale experimental power plants -- each with an additional outer containment building and machinery to absorb radioactive iodine? Perhaps we should opt for nine experimental power plants. With eight nuclear power plants we can run a full factorial of three factors, each at two levels. With nine we could accommodate a Graeco-Latin square with three levels for four factors; however, we could not estimate interactions. Neither design would provide a clean estimate for error; therefore should we double these numbers to achieve replication in each cell?

The last few sentences have been deliberately couched in statistical jargon and are facetious in this context. However, they are valid teasers. What constitutes plausible evidence (never mind scientific or compelling evidence) as a basis for regulatory action? How much should a probabilistic risk assessor know about statistics and the principles of design of experiments-even if he does not conduct experiments?

## 3. A Personal View on Uncertainty in Nuclear Regulation

The Post Lewis idea that we will substantially advance nuclear regulation if we just quantify the uncertainties (and have peer review) is in my opinion more a matter of illusion than substance. The tautology "If one does not know, one does not know" has obviously a much firmer basis, and, in my opinion, has more relevance to regulation. I do not believe that we can quantify uncertainties in a meaningful way over the whole range of regulatory problems. I find the following proposition difficult to accept as a general rule. We may not understand a phenomenon very well and are therefore uncertain about it; yet at the same time we understand it and our process of thinking about it so well, that we know the nature of the error in our thinking and therefore can quantify the uncertainty. I find empirical confirmation of my somewhat philosophical probing in an occasional gesture by some NRC engineers. The arm is raised with an open hand; the arm is pulled down and the hand is closed - the value of interest and its uncertainty was pulled out of the air.

I have not undertaken a review of studies on uncertainty. In his testimony, Okrent cites a study of his [2] which at least in some respects is similar to what I read or heard elsewhere. In his testimony, Okrent summarises that "... seven [respondents, i.e.] seismologists and geologists making their judgments independently, usually differed by a factor of $10^{-3}$ to $10^{-4}$ in their estimates of return frequencies for wide range of earthquakes at eleven different reactor sites." Reference [2] also deals briefly and rather vaguely with respondents' assessment of their uncertainty. While it is not clear what they understood by uncertainty (standard deviations (of what random variable or population), maximum error, or whatever*), from a pragmatic point of view their estimates of the occurrence rates and their assessments of the "uncertainties" in their estimates are not consistent. Of two respondents one "...generally estimated an uncertainty of 10-20%", and another, "...of a factor of two in the probabilities per year." However, on several estimated return frequencies they differed by as much as a factor of 1000. Other respondents' (vaguely stated) estimates of uncertainty also do appear too small in terms of the spread of the

* Are they personal uncertainties, or are they in some sense objective? Are they dependent on specific theories which in turn may not be firmly established? Were such theories shared by several or all assessors?

estimated probabilities among all respondents.

While the case study of Okrent may be a rather extreme example, I believe that on many NRC problems

a) the uncertainty is large - one, two, or perhaps even three or more orders of magnitude:

b) the nature of the uncertainty is vague

c) whatever the conceptual basis of uncertainties they are misestimated from a common sense point of view and often underestimated or grossly underestimated.

Often these large and not well understood uncertainties need to be combined and propagated into a "final" uncertainty with no compelling prescription for combining and propagating uncertainties. Frequently one would expect that in some sense the uncertainty of the final result is larger than that of any of its inputs and therefore very large and that points (b) and (c) above are also amplified. Let me combine all these aspects of uncertainty under the label uncertainty complex.

There are many more or less specific facets of uncertainty that merit consideration. Later on I shall deal with some psychological aspects of analysis which relate to uncertainty. For sake of brevity I shall deal with only one more aspect under the label of futurology. The primary purpose of risk analysis is to assess future risk. While it is indeed reasonable to project from the past and present such projections are not error free. Reliability growth is a very plausible effect and most likely the dominant one. However, there are also potential adverse changes the likelihood of which is speculative.

1) Bathtub curve: most of the reliability data is from commercial reactors less than 20 years old. There is a possibility that some failure rates of vital components or systems could rise sharply after 20 years.
2) State of emergency plans 5 to 10 years hence.
3) A presumed safety feature backfires.
4) Sabotage from dissatisfied labor or terrorists.
5) Several years of successful operation bringing about complacency.
6) Economic conditions promoting shortcuts.

This list could probably go on; quite possibly if a very detrimental change were to come it might not be even thought of now.

There are various ways of viewing the uncertainty issue in nuclear regulation. Okrent in his first sentence says "...(NRC) makes decisions concerning the public health and safety in the presence of very considerable technical uncertainty." My reaction is that the NRC decision process is beset with an overwhelming uncertainty complex. It brings to mind the emperor without clothes. This must be an uncomfortable position for NRC as it would be for any regulating agency. However, it is not a circumstance about which NRC needs to be apologetic. We are in a new domain with many phenomena about which dependable knowledge has not been obtained. As Okrent points out in his introduction many regulatory agencies are regulating under similar circumstances. In fact, uncertainty complexes beset our lives as individuals as well as collectively as a nation. They range from difficulties in raising children to problems of national defense. The latter may affect the likelihood of nuclear war which in comparison would make any nuclear power plant catastrophy look puny.

If my assessment of the NRC uncertainty complexes is correct, then NRC has three broad choices.

a) It can take the current type of risk analysis at face value, make regulations in accordance with their results, and bluff on the validity of its decisions.
b) It can start or continue with vigorous efforts to make the probabilistic risk analysis more rigorous and convincing.
c) It can explicitly acknowledge profound uncertainty and regulate with recognition of this limitation.

These options are not mutually exclusive. One can use various shadings of these options, and the shadings may vary with the circumstances as indeed ~~is~~ the case now. Superficially option (b) appears attractive. However, I believe that in terms of "reasonable" precision many of our problems are intractable and will continue to be so. I would expect only a slow nibbling away at a problem here or there. Option (a) does have the advantage that in principle it maintains a stronger degree of authoritativeness than option (c). Undoubtedly it is tough to regulate without an air of authoritativeness. However, option (a) may lead to bad decisions and may not be viable in the democratic process in which NRC must function. This brings us to facing up to option (c). In a fashion option (c) was or is operative for persons who favor the judgmental approach to regulation. Option (c) in my view does not necessarily call for judgmental approach. The intended thrust of option (c) is that both judgment and probabilistic risk assessment are very limited tools for assessing nuclear risks or for optimizing benefits against penalties with respect to nuclear energy.

### 4. Some Additional Discussion of Analysis and Judgment

I do not attempt here to resolve what role a highly fragile risk analysis should play in regulation, and in particular its relationship to or in competition with judgment.* This subject matter is outside my area of competence. I can only present some ideas related to it and state what my inclinations are. Before doing that I shall summarize some points that weaken the case for probabilistic risk analysis.

Particularly in difficult problems, mathematical analysis provides its own straightjacket. The analyst will only use methods he or she knows and that do not require an inordinate amount of effort. Thus certain types of failures will be treated as independent, constant failure rates will be postualted, or generic values will be applied to differing members of a class. Even if the analyst is sophisticated enough to use one of the few models for dependence, he or she is still limited to the few known models, all of which have limitations. The phenomena of nuclear power plants are very complex; many cannot be dealt with realistically with workable models. The thought processes of the analyst may in part be dictated or influenced by the medium of his choice; i.e., the mathematics that is practically available to him. This applies to the best analyst as well as the poorest; even though the former can deal more deftly with limitations and will generally

* C. Bennett and M. Ernst pointed out that it is the relationship or interaction between analysis and judgment that is paramount; I agree. Nevertheless, there are differing views on the relative reliance to be placed on each.

have more awareness of the limitations. The straightjacket idea is illustrated by the rather frequent and sometimes unjustified criticism directed at the mathematical analyst: "You have a beautiful solution to the wrong problem."

Analysts differ in their awareness of the limitations of their analyses and in their efforts to report that awareness. Some become so engrossed with their achievements that they do not see the shortcomings. Others while perhaps recognizing the shortcomings may not report them; in fact some may present their analysis with puffery. Perhaps the majority will state briefly and rather inconspicuously some of the assumptions and limitations of the analysis. It is my impression that only a minority of analysts at NRC drive home with force and proper elucidation the limitations of their analysis. The analyses are provided directly or indirectly to "users" which may be colleagues, supervisory personnel, the commissioners, or ultimately the public. Even if limitations of the analysis are serious and stated forcefully, the user has a strong tendency to take the results at face value - particularly if they confirm his predilection, or seemingly help him to get out of a dilemma. Analysis no matter under what label (mathematical, statistical, risk) has a ring of authority and authenticity. It is often unquestioningly accepted by the lay analyst. In fact even capable analysts are affected by the halo effect of "analysis". Unless

a) they give other persons analysis careful scrutiny

b) had experiences with the type of analysis under review,

they might accept the results with less reservations than they deserve.

Besides the wrong psychological impact, analyses often have unjustified staying power. Early analyses become the basis of later analyses, thus relieving the later analyst of having to deal with the tough issues of not well understood phenomena. It is much easier to cite than to investigate and think through difficult problems. Individuals and institutions will defend their analyses and insist they are valid, discounting evidence to the contrary. While I sense in NRC lessening rigidity in the defense and use of past analyses, the problem of unjustified staying power is still with us.

If formal analysis has so many limitations-I do not claim a complete catalogue of them - are the results of judgment to be trusted more than those of formal analysis? As I indicated earlier I lack competence to answer this question. For the sake of discussion let me speculate. It is conceivable that a knowledgeable person with a subtle mind might provide better answers for the following reasons. He or she:

a) is not bound by the formalism of analysis
b) brings to bear conscious and unconscious knowledge and wisdom; and
c) has a broader perspective on the problem than the formal analyst.

On an intellectual level I am not convinced by such speculation, and even if true in some or most cases, how do we decide which are these cases? Also, how do we decide who is the most knowledgeable, wise, and subtle person to provide this judgment? Yet on an emotional level, I tend toward judgment over analysis in nuclear regulation for the following reasons:

1) Too much credibility is given to analysis.
2) The staying power of analysis is too strong.
3) The judgment and its limitations are often more readily understood than the analysis.

While some of the same causes for points (1) and (2) also function for judgment, I believe that they function less intensively for judgment. I believe that judgment will in general be accorded less unjustified authority and less staying power than analysis.

Okrent states "Criteria for judgment should be developed...". NRC has difficulty developing criteria for good analysis. I believe it is even more difficult to develop criteria for good judgment. Despite the bleak outlook, such an effort may be worthwhile. I believe it should be combined with a review of the philosophy of regulation, in particular with option (c), namely that NRC explicitly acknowledge profound uncertainty, and regulate with recognition of this limitation. The review group should have profound thinkers and good pragmatists (not necessarily mutually exclusive). It could conceivably be supported by several regulatory agencies having common problems.

## 5. Postscript

It might be inferred from the text above that I believe that probabilistic risk analysis is useless, or nearly so, for regulatory purposes. This is not my point at all. I expressed my concern with various limitations of analysis and its misuse particularly with respect to uncertainties. I do believe that probabilistic risk analysis does have a vital role to play in NRC regulations. The why and how of this role I prefer not to tackle in this document. Some issues of risk analysis in NRC are raised in my talk to the ASA Ad Hoc Advisory Committee on Nuclear Research (3).

## 6. Acknowledgments

Dr. Carl Bennett, Mr. Malcolm Ernst, Mr. William Maher, Mrs. Esther Rubinstein, Mr. Joel Rubinstein, and Mr. Norman Wagner provided constructive criticism. I thank them for their help.

## 7. References

1. David Okrent, Testimony to the Nuclear Safety Oversight Committee, January 20, 1981, Santa Barbara.

2. A Survey of Expert Opinion on Low Probability Earthquakes, Annals of Nuclear Energy, Volume 2, pp 601-614, 1975.

3. David Rubinstein, A Statistician's View of NRC Statistics, edited transcript of talk to the ASA (American Statistical Association) Ad Hoc Advisory Committee on Nuclear Research, November 7, 1980, Washington, D. C.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

3/18/82

NOTE TO: Frank Coffman

FROM: Jim Conran

SUBJECT: COMMENTS ON DRAFT LETTER (HANAUER TO COOPER-NUPPSCO)...AND RELATED MATTERS.

The Draft letter looks ok to me as is; but a bit more should be said about the pot that is boiling here. I have talked to Carl Johnson, RES (NRC's representative on the NUPPSCO Committee referred to in the incoming) regarding the safety classification/terminology issue involved here. He sent to me the draft of NUPPSCO's proposed "alternative definitions" (see attached) plus a Minority Report reflecting the views of dissenting members of the NUPPSCO Committee (also attached). My comments to him regarding these matters were:

(a) The proposed definitions may be consistent (as claimed) with Denton's "standard definitions" (approved by Denton on 11/20/81 - see attached); but it would surely involve a very substantial review effort to demonstrate/prove that point. And even if that were done, in my opinion we would not have "gained" anything; we would only then have additional, new safety classification terms which we would then have to try to get everybody to learn and use consistently.

(b) We really don't need any new (alternative) safety classification terms defined; we just need standardization (consistency), within both the staff and the industry, in the usage of the terms already included in the regulations and existing regulatory guidance document (e.g. Reg. Guides, SRPs, NUREGs, etc.). That was the purpose of Denton's 11/20/81 memo to all NRR people. The need to take the next obvious step (i.e. incorporating Denton's standard definitions into the "DEFINITIONS" sections of the regulations so that staff and industry must/can use them consistently) is readily apparent from the NUPPSCO dissenter's usage of the term "Important to Safety" in the Minority Report.

(c) Although I am not, and we (NRR) should not be, receptive to the proposed new "alternative" safety classification language, the underlying or associated industry effort to understand the relative safety importance of reactor plant components, and to establish a basis for sorting those components into various categories, should be of great interest to us. I think Hanauer is right in wanting to talk with industry about their approach, categorization bases, etc. This more interesting and potentially useful aspect of the industry effort in developing the new proposed standards ANS 51.1 and ANS 52.1, is apparently spelled out in considerable detail in those draft standards; so I have asked Carl to obtain and send to us copies of them prior to the (rescheduled) Hanauer meeting with NUPPSCO members. That kind of info is clearly related to,

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ENCLOSURE 6

and would probably be valuable input to a number of RES and NRR "importance ranking" type efforts already underway (e.g. A-1297 (RRAB); A-1295 (RRAB); Walt Haass' Graded Q.A. Development effort just getting underway with EGG as contractor; RES long-term importance ranking/graded Q.A. effort just getting underway with Sandia as contractor; etc.).

Our (RRAB and DST) proper course for the future regarding this general topic should be to propose that Denton's "standard definitions" be formally incorporated into the appropriate "DEFINITIONS" section of the regulations (e.g. 10 CFR 50 Appendix A and Appendix B, and 10 CFR 100 Appendix A). This would clarify to the public and the industry (as well as to the entire NRC staff) that we (NRC) know what our regulations and regulatory guidance mean , and that we intend to enforce consistent interpretation and application of them. At the same time, we must be sensitive to the industry's concern (as reflected in the NUPPSCO Minority Report) that by clarifying and insisting on consistent usage of the language of our regulations, we are "changing the meaning" of that language (e.g. important to safety) in order to sneakily ratchet or broaden the scope of the existing regulations. For that reason, the same language Denton used in his 11/20/81 letter to ALL NRR to emphasize that point should be included in the "Discussion" section of the Proposed Rule that would incorporate his "standard definitions" as I have suggested.

To really wrap this thing up right, we should also initiate the development of another Reg. Guide and another SRP section to provide further detail/discussion/ guidance to both the staff and the industry regarding proper application of Denton's "standard definitions." I know that Thadani and Ernst have been somewhat reluctant to involve us heavily in this kind of activity in the past because of our severely limited resources; but the passage of time has indicated clearly, I believe, that if we (who happen to know best the "background" of the development of Denton's "standard definitions") don't take the initiative in getting done what I am recommending, it simply isn't going to get done. And, as you know, a great deal of support has developed for getting it done (e.g. from ELD, RES, ACRS, and ASLB, TMI-1 Board) as a result of our having addressed these issues in a number of different contexts over the past 1½ years. This is not just "word smithing"; what is involved is the precise meaning of the specific language that describes some of the most fundamental concepts of our regulatory structure and philosophy. We really ought to get it (consistently) right, sometime soon.

Jim

Jim

cc: J.Conran Chron

# ROUTING AND TRANSMITTAL SLIP

Date 7/18/82

| TO: (Name, office symbol, room number, building, Agency/Post) | Initials | Date |
|---|---|---|
| 1. Frank Coffman | | |
| 2. Ashok Thadani | | |
| 3. Mel Ernst | | |
| 4. Steve Hanauer | | |
| 5. | | |

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| | Approval | | For Clearance | | Per Conversation |
| X | As Requested | | For Correction | | Prepare Reply |
| | Circulate | | For Your Information | | See Me |
| X | Comment | | Investigate | | Signature |
| | Coordination | | Justify | | |

REMARKS

I would like to request that the personal views expressed in my note be sent forward with the concurrence package as it goes back up the line. Specifically, I would like to raise again for DST/RRAB management reconsideration the question of whether we should not be taking more initiative in the course of action recommended.

JHC

DO NOT use this form as a RECORD of approvals, concurrences, disposals, clearances, and similar actions

| FROM: (Name, org. symbol, Agency/Post) | Room No.—Bldg. |
|---|---|
| Jim Conran | 310 |
| | Phone No. 2-7111 |

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OPTIONAL FORM 41 (Rev. 7-76)
Prescribed by GSA
FPMR (41 CFR) 101-11.206

☆ U.S.GPO:1977-0-241-530/3238

EXCERPT FROM

REBUTTAL TESTIMONY ON CONTENTION 7B, DATED 7/1/82,
BY. J. H. CONRAN TO THE SHOREHAM ASLB

A. The Staff does not believe it is acceptable for the language differences indicated in the statements on p. 55 of Applicant's testimony to go unresolved because of certain unacceptable implications of the different usage of the safety classification language of the regulations. These implications obtain not only with regard to Shoreham licensing but also with regard to the efficacy of the Staff's approach and methods of safety review in more general application. There are at least three such implications:

1. Because the Staff conducts an audit review, reliance must be placed on commitments by Applicants that all portions of the regulations are complied with (see, e.g., FSAR § 3.1.2.1). It is critical that these commitments mean what the Staff understands them to mean if the Staff's determination of "reasonable assurance" (which finding must be made in accordance with 10 C.F.R. § 50.35(c) in order to license a facility) is to be meaningful in the sense intended in the regulations.

2. It is clear under the Staff's understanding of "important to safety" (but not under Applicant's) that there exists in the regulations a requirement under GDC 1 for a QA program for certain non-safety-related structures, systems and components (i.e., those important to safety).

3. Under Applicant's construction of "important to safety," the obligations imposed by 10 C.F.R. Part 21 might be more narrowly construed than would be the case under the Staff's broader definition of that term.

These examples demonstrate why agreement on the safety classification definitions provided by the Denton definition is extremely significant.

* U.S. GOVERNMENT PRINTING OFFICE: 1983-381-299:133

1. BOARD NOTIFICATION (82-98) REGARDING QC REQUALIFICATION PROGRAM DTD 9/28/82

2. BOARD NOTIFICATION - WELDS IN MAIN CONTROL PANELS (BOARD NOTIFICATION NO. 82-90)

3. RECOMMENDATION FOR NOTIFICATION OF LICENSING BOARD (to Eisenhut from Warnick 11/1/82) (Also filed in reading files)

4. RECOMMENDATION FOR NOTIFICATION OF LICENSING BOARD (to Eisenhut from Warnick 12/3/82)

5. RECOMMENDATION FOR NOTIFICATION OF LICENSING BOARD (to Eisenhut from Warnick 12/01/82)

6. INFORMATION ITEM - NOTIFICATION OF WORK STOPPAGE ON HVAC WELDING AND MAJOR REDUCTION IN OTHER SAFETY-RELATED WORK (BN-82-126) (to ASLB from Novak 12/07/82)

7. BOARD NOTIFICATION - ALLEGED DESIGN DEFICIENCY (BOARD NOTIFICATION NO. 82-105)

8. BOARD NOFITIFCATION NO. 82-123 - USGS OPEN FILE REPORT ON PROBABILISTIC ESTIMATES OF MAXIMUM ACCELERATION AND VELOCITY IN ROCK IN THE U.S.

9. BOARD NOTIFICATION - ACRS EVALUATION OF PWR FLOW BLOCKAGE (BOARD NOTIFICATION NO. 82-125, 82-125A)

10. BOARD NOTIFICATION - USGS POSITION ON THE CHARLESTON EARTHQUAKE (BOARD NOTIFICATION 82-122A)

11. INFORMATION ITEM - APPARENT DEFICIENCIES IN MIDLAND-ROSS "SUPERSTRUT" MATERIAL USED FOR CLASS 1E CABLE TRAY AND CONDUIT SUPPORT (BOARD NOTIFICATION 83-02)

Wayne - Are we following this? Are we tracking it · the OII? ~~[illegible]~~ ② File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Ahearne
Commissioner Roberts
Commissioner Asselstine

FROM: Darrell G. Eisenhut, Director
Division of Licensing, ONRR

SUBJECT: INFORMATION ITEM - APPARENT DEFICIENCIES IN MIDLAND-ROSS "SUPERSTRUT"MATERIAL USED FOR CLASS 1E CABLE TRAY AND CONDUIT SUPPORT (Board Notification No. 83-02 )

In accordance with present NRC procedures for Board Notifications, the enclosed information is being provided for your information. This information may be applicable to all nuclear power plants.

The enclosed memorandum discusses the lack of adequate quality assurance on "superstrut" material used in cable tray, conduit, and instrument supports. Diablo Canyon Units 1 and 2, Palo Verde Units 1, 2 and 3 and Washington Nuclear Project, Units 1 and 4 have been identified as having used this material. The staff is reviewing the safety implications of this matter and will promptly notify you of any significant developments. At this time the applicability of this issue cannot be limited to these three facilities. Therefore, all Boards are being notified according to NRR procedures. When we have evaluated the individual or the generic implications of these findings, we intend to notify all appropriate parties.

Darrell Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Bob
first notification
We will track as necessary
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Enclosure:
As Stated .

cc: See Next Page

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cc: OGC
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SECY
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The Atomic Safety & Licensing Boards for:

Callaway Plant, Unit 1
Clinton Power Station, Units 1/2
Comanche Peak Steam Electric Station, Units 1/2
Diablo Canyon
Midland Plant, Units 1/2
Palo Verde Nuclear Generating Station, Units 1/2/3
Shoreham Nuclear Power Station
South Texas Project 1/2
Waterford Steam Electric Station, Unit 3
William H. Zimmer Nuclear Power Station, Unit 1, and

The Atomic Safety & Licensing Appeal Boards for:

Comanche Peak Steam Electric Station, Units 1/2
Diablo Canyon Nuclear Power Plant, Units 1/2
Enrico Fermi Atomic Power Plant, Unit 2
San Onofre Nuclear Generating Station, Units 2/3
Virgil C. Summer Station, Unit 1
Waterford Steam Electric Station, Unit 3
William H. Zimmer Nuclear Power Station, Unit 1
FNP 1-8