UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD
Before Administrative Judges:
James P. Gleason, Chairman
Frederick J. Shon
Dr. Oscar H. Paris

In the Matter of	)	
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point, Unit No. 2)	)	Docket Nos. 50-247 SP 50-286 SP
POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	)	Jan. 17, 1983

LICENSEES' TESTIMONY OF DENNIS C. BLEY AND DENNIS C. RICHARDSON ON CONTENTIONS 2.1(a) AND 2.1(d)

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TESTIMONY ON
FILTERED VENTED CONTAINMENT SYSTEM
AND SEPARATE CONTAINMENT SYSTEM

A. Introduction

My name is Dennis C. Bley. I am a consultant at Pickard, Lowe and Garrick, Inc. in reliability, risk, and decision analysis for electrical generating plants. I was a principal investigator on the Indian Point Probabilistic Safety Study. A statement of my professional qualifications is attached.

My name is Dennis C. Richardson. I am the Risk Assessment Technology Manager in the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation. I was a principal investigator on the Indian Point Probabilistic Safety Study. A statement of my professional qualifications is attached.

We have read the testimony of Messrs. Gordon R. Thompson and Steven C. Sholly addressing Commission Question 2 and Board Contention 2.1(a) (filtered vented containment system (FVCS)) and 2.1(d) (separate containment system (SCS)) and have substantive causes for rejecting their line of reasoning that concludes that "implementation of a filtered vented containment system or a compartment venting system is necessary at Indian Point Units 2 and 3." UCS/NYPIRG Testimony of Gordon R. Thompson and Steven C. Sholly on Commission Question Two, Contentions 2.1(a) and 2.1(d) at 19 (Dec. 28, 1982).

We will show that Messrs. Thompson and Sholly are incorrect in claiming that "to achieve significant risk reductions, proposed solutions must address accident consequences." Id. at 19. Either preventive measures or mitigative features can reduce risk. Thompson/Sholly also fail to show that the risks from the Indian Point plants are unacceptable and that either preventive or mitigative features are warranted. By citing documents which were published prior to the issuance of the Indian Point Probabilistic Safety Study (IPPSS), inappropriately using the containment failure modes identified in the Reactor Safety Study (WASH 1400), and applying generic concepts to plant-specific cases without a plant-specific assessment, Messrs. Thompson and Sholly overstate the feasibility, overstate the risk reduction capability, understate the costs, and deemphasize the potential problems with such devices.

A plant-specific, site-specific safety study, the IPPSS, has been performed to ascertain the risk from operation of Indian Point Units 2 and 3. This study was issued in March, 1982, and has received extensive peer review. It provides essential information for evaluating Board Contentions 2.1(a) and 2.1(d). As will be testified later in this proceeding under Commission Question 1, the risks of operation of these units are very low. The societal and individual risks fall well within the safety goals regarding off-site consequences just adopted provisionally by the Commission. The results of the IPPSS demonstrate that the dominant scenarios identified in the Reactor Safety Study, such as early containment overpressurization, and uncritically assumed by UCS/NYPIRG to be important for Indian Point, are not applicable to the Indian Point plants.

Addressing key aspects of the Thompson/Sholly testimony, we agree that the risk to the public from the operation of Indian Point Units 2 and 3 is dominated by core melt accidents. This is so because it is only such accidents which provide even a theoretical mechanism for releasing a large fraction of the radioactive inventory from the core. Their statement that "[d]espite the considerable efforts taken . . ., core melt accidents dominate risk," _id._ at 5, ignores this simple fact.

Under the assumption that the risks must be reduced, Thompson/Sholly dismiss the effectiveness of efforts to lower risks by decreasing accident sequence frequencies. They argue for this point by citations to inapplicable references, _i.e._, assuming generic studies are completely applicable to the Indian Point plants. At the same time, they suggest that the FVCS and SCS features would delay core melt by two days. They are wrong on both counts. The sequences for which the mitigative features would delay core melts do not apply to the Indian Point plants. On the other hand, effective measures do exist for reducing core melt frequencies. In particular, we have pointed out in our testimony regarding the Director's Confirmatory Order[1] that the IPPSS identified risk contributors amenable to improvement by plant modifications. Those cost-effective changes are being implemented voluntarily by the licensees and yield substantial improvement in risk. No such cost-effective gains have been demonstrated for the FCVS and SCS.

Lastly, the Thompson/Sholly testimony makes assumptions about the quantity of radioactive material which would be released to the environment under the postulated accident sequences which it considers. Their conclusions as to the achievable benefits from the FVCS and SCS directly depend upon such source terms. No analysis of such devices can be valid without a critical assessment of source terms, which intrinsically affect results of any evaluation of the potential benefits.

B. Capability of the Indian Point Containments

A better understanding of the capabilities of the present Indian Point containments is necessary in order to evaluate the UCS/NYPIRG claims. While Question 1 will address containment capability in detail, several facts are significant to this testimony. These 2.6 million cubic foot structures are strong enough to withstand the largest earth-

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1. Licensees' Testimony of Dennis C. Bley and Dennis C. Richardson On Commission Question 2 and Board Question 2.2.1 (Jan. 12, 1983).

quakes that can be reasonably postulated for this area.

Another measure of the strength of these containments is the very high internal pressures that would have to be achieved before containment integrity would be lost. Although it has always been recognized that the actual failure pressure was considerably in excess of the design pressure, no specific detailed analyses had been performed for the Indian Point containments until the IPPSS analysis. This analysis calculated the pressure limit to be about 141 pounds per square inch absolute (psia), some 2.3 times higher than the design pressure. The structural analysis methods used for analyzing the Indian Point containments accurately predict structural behavior when applied within the buildings' elastic limits. The structural criterion used for defining the Indian Point failure pressure is the onset of yielding (i.e., the point at which materials begin to deform beyond the elastic limit) at the most limiting locations.

The IPPSS containment capability analysis is well supported by other independent analyses. NUREG-0850 reports an initial calculation of the Indian Point containment failure pressure of 133 psia and anticipates that refined analyses would raise this value to 148 psia. Additionally, the NRC Staff in its testimony cites a containment failure pressure of 141 psia. Direct Testimony of James F. Meyer Concerning Contention 2.1(a and d) at 3 n.* (Jan. 12, 1983).

Such high containment capability has the following effects on the risk. First, the IPPSS has determined that pressure surges which would lead to an early containment failure due to potential steam spikes or hydrogen explosions occurring singularly or concurrently will almost always be below 140 psia at Indian Point. Early containment overpressurization is therefore not a major contributor to the risk of either early or latent health effects.

Second, assuming a core melt and a total loss of containment heat removal capability, it would take a fairly long time for pressure conditions to build up in the containment to reach the failure point. The IPPSS demonstrates that delayed overpressure failures of this type do not contribute to early fatalities. It was conservatively assumed in the IPPSS that containment overpressure failure would occur about 12 hours after accident initiation. This represents a significant conservatism because it is more likely greater than about one day after the initiating event before the 140 psia level could be reached.

Much can be done in this time, including operator action to terminate the pressure rise altogether; evacuations could be implemented, if required; short-lived isotopes would decay away. If no more than one containment fan out of the five installed fans were restored, or one out of the six containment spray modes were made functional, or other ad-hoc recovery measures taken, the containment could maintain its integrity. The already low failure probability of the Indian Point containments is conservatively estimated in the IPPSS in that these recovery actions following core degradation were not quantified. If minimum containment heat removal capability either existed throughout the accident sequence or was restored during the recovery period, failure pressures should not be reached and public health consequences from the accident will be negligible.

C. Core Melt Frequency Is a Poor Indicator of Risk

In the Reactor Safety Study, detailed containment capability studies of the type in the IPPSS were not done and containments were assumed to fail at a pressure significantly below that calculated in the IPPSS. The plant analyzed in the Reactor Safety Study was not representative of the Indian Point units and adequate consideration was not given to the detailed phenomena which would be associated with core melt. Therefore, core melts were closely coupled in this earlier study with containment failure and, thus, radiological risk.

Our understanding of risk levels has changed significantly because of the advances made in the analyses of the Indian Point containment capability and our consideration of the phenomenology which would be associated with core melt at the specific Indian Point plants. We now realize that risks from these plants are even lower than previously believed. Contrary to the UCS/NYPIRG testimony, core melt frequency -- once thought a good indicator of radiological risk -- is, in fact, a poor indicator at Indian Point. Instead, risks from the Indian Point plants are determined by a few dominant scenarios. For example, more than 90 percent of the early fatality risk at Indian Point is due to one sequence, the interfacing systems LOCA. This sequence would also bypass containment and would, therefore, also bypass UCS/NYPIRG's proposed FVCS and SCS. Yet, this sequence contributes less than one-half of one percent to the overall core melt frequency.

Many of the studies relied upon by UCS/NYPIRG are based on Reactor Safety Study-type containment failure assumptions found in the IPPSS to be incorrect for the Indian Point units. Reliance on such studies is, therefore, seriously flawed. Correspondingly, the reduction factors claimed by UCS/NYPIRG for their FVCS and SCS are based on outdated information. Whereas UCS/NYPIRG refer to a reduction in the early fatality risk to just 8 percent of its present value (page 16, A.24, Thompson/Sholly Testimony), FVCS and SCS are ineffective for bypass scenarios and would not reduce the early fatality risk at Indian Point.

D. Comparative State of Knowledge of the Large Pressurized Water Reactor (PWR) Containment Versus FVCS and SCS

The state of knowledge of the design and performance of large PWR containments far exceeds that of FVCS and SCS. Part of the extensive knowledge of large PWR containments stems from the hundreds of reactor years of day-to-day operating experience and the experience gained from periodic pressure and leakage tests. The IPPSS analysis and other recent dry PWR containment evaluations have added to this knowledge.

The UCS/NYPIRG mitigation concepts involve modification of the Indian Point containment boundary. To the best of our knowledge, there are no filtered vented containment systems or separate containment systems in commercial nuclear power plants in the United States. Although UCS/NYPIRG has referred to Barseback, which is a boiling water reactor, the applicability of this concept to the Indian Point large dry containments is unknown without a plant specific analysis. Similarly, the relevancy of a proposed filtered venting system for the Clinch River Breeder Reactor has not been established by UCS/NYPIRG (page 16, A.23, Thompson/Sholly Testimony). It is difficult to understand how a proposed design for a liquid metal cooled reactor, for which only a few details of the filtered vent are available, would form a persuasive argument for altering the Indian Point plants. As to a separate containment structure, UCS/NYPIRG did not identify any specific plant where this concept is even being considered.

Neither of these devices has been constructed and no operating experience exists. We cannot know the true value of their specific proposals when no details are provided. It appears that UCS/NYPIRG has not carefully evaluated the worth of these systems either, for they quote NUREG/CR-0165, NUREG/CR-0138, and SAND 80-0887 for reductions in early fatalities due to filtered vents. Given our current state of knowledge, they do not apply to the Indian Point plants. As stated previously, for early containment failures, FVCS would not reduce early fatality risk at Indian Point.

E. Potential Benefits and Detriments of FVCS and SCS

UCS/NYPIRG does not provide plant-specific probabilities and overstates the reduction in health effects associated with these proposed systems. Although they would have no effect on early fatalities due to early containment failures, FVCS and SCS might reduce latent health effects and property damage. However, the Indian Point plants already meet the early and latent fatalities safety goals adopted by the Commission. The FVCS and SCS do not prevent accidents; nor do they reduce the frequency or types of accident initiators or of core melt.

The principal health effect where the FVCS and SCS might have a role to play is in reducing the latent fatality risk. Certain severely demanding design criteria must be met or these features may fail to realize even this limited role or indeed might even contribute to increasing the consequences of an accident. The FVCS and SCS must be capable of withstanding all of the present initiating events which dominate the latent fatality risk in such a manner as to preclude failure modes that could cause an early release of radioactive material. For example, the present pressure boundary of the Indian Point containments will not fail from any credible seismic event. Similarly, the pressure boundary of the FVCS and SCS must not fail from any credible seismic event, otherwise a potentially late overpressurization scenario could become an early release scenario. Such a failure would cause not only latent fatalities and property damage, but also add early fatalities. Similar logic must be applied to fire and wind initiated scenarios which, with seismic scenarios, now influence the latent fatality risk at Indian Point. Assuring that this criterion is met requires careful and detailed analysis which would have a large impact on the potential costs of these systems. The best approach to this is to perform a probabilistic risk assessment on a specific design as was done in the IPPSS.

In order to properly evaluate in a quantative manner any proposed hardware change to a plant, including the FVCS and SCS, probabilistic risk assessment techniques should be utilized. This was in fact clearly intended in the Commission's Orders establishing this proceeding. "Risks from nuclear power reactors are defined by the probabilities and consequences associated with potential accidents. . . . Despite these uncertainties [associated with quantitative risk assessment calculations], risk assessment methods offer the best means available for objective and quantitative comparison of the kind needed here." Memorandum and Order at 8 (Jan. 8, 1981).

One part of this technique is the use of the containment event tree which provides a thorough and structured study of the effect of the system or system changes on the progression of the core damage events and containment integrity. For each proposed feature, modifications should be made to the split fractions where the feature would affect the outcome of the containment event tree node (either positively or negatively). These split fractions should then be used to establish a revised containment matrix for each mitigating feature. These modifications must then be propagated through the risk calculations. Comparing the family of risk curves thus calculated with the base risk identifies the risk reduction which the addition of the feature will afford.

This methodology is the manner in which the effect of a mitigation system on risk should be quantified. This procedure should include the following:

- Indian Point plant specific accident initiation frequencies.
- The inclusion of a specific study of the effect of operation, maloperation or failure of the mitigation system on the probability of reaching different degraded core situations.
- The effect of the mitigation system operation or misoperation on the containment failure probability.
- The effect of the mitigation system operation or misoperation on the timing of and type of radiological release.
- The inclusion of the above items in a detailed assessment of the probability of and consequences of core meltdown accidents at Indian Point.

Without performing this detailed plant specific assessment for internal and external events, the risk reduction of a mitigation system for the Indian Point units cannot be quantified. In fact, without working with a plant specific risk study (i.e., the IPPSS), the risk reduction cannot even be estimated.

Before performing this detailed and costly assessment of a mitigation system, three questions must be considered;

- First, is the analysis necessary, considering the low risk already present at the Indian Point plants and the potential for further reductions in estimates of risk due to source term research?
- Second, does the technology exist?
- Third, are the costs appropriate?

If these systems were backfitted to the Indian Point plants, a detailed specification of design criteria and performance requirements would be necessary. These specifications would have to account for a wide variety of conditions under which the systems may be implemented or called upon for use, including flow rates and steam/air/hydrogen compositions which cover a wide range of postulated accidents. A preliminary checklist of design requirements as set forth below was taken from NUREG/CR-1410, referenced by UCS/NYPIRG. However, the items on this list, which indicate the complexity of designing such systems, were not discussed in UCS/NYPIRG's testimony.

<u>Checklist From NUREG/CR-1410</u>

1. Functional Requirements
   a. Containment pressure reduction
   b. Containment temperature reduction, if necessary
   c. Mitigation of radioactive release
2. Operational Requirements
   a. Decontamination factors for important isotopes
   b. Quality assurance criteria (especially for sand filters)
   c. Maximum filter loading capacity and fission product re-entrainment characteristics
   d. Maximum and minimum flow rates and pressure drops
   e. Heat removal and condensate drainage requirements

f. Capability to withstand operating environment

g. Instrumentation requirements

3. Resistance to Hazards

a. Resistance to earthquakes, tornadoes, and missiles

b. Resistance to fire and hydrogen explosions within filter system

c. Resistance to steam explosion from within containment

4. Reliability

a. Valve actuation reliability

b. Reliability of mechanical components (air coolers, hydrogen recombiners, and heat exchangers)

c. Likelihood of spurious operation

d. Likelihood and impact of human error

e. Filter failure or bypass modes and likelihood of occurrence

f. Emergency power requirements

g. Redundancy

5. Control

a. Actuation logic

b. Flow rate control

c. "Zero-release" options

6. Sabotage Protection

a. Passive operation versus operator control

b. Protection of piping, valves, and filters from unwanted access

7. Inspection Considerations

   a. Ease of access

   b. Frequency of inspection

   c. Inspection objectives:

      (1) Evidence of structural damage or degradation

      (2) Water infiltration, weathering

      (3) Contamination with foreign matter

   d. Impact on plant operating procedures

8. Testing Considerations

   a. Frequency of testing

   b. Testing objectives:

      (1) Efficiency degradation versus time

      (2) Flow resistance versus time

      (3) Component availability

   c. Testing methods

9. Maintenance Considerations

   a. Ease of access

   b. Periodic replacement of filter materials (especially charcoal)

   c. Grooming of filters (especially sand bed)

10. Post-Accident Safety and Repair-Restoration Considerations

    a. Shielding criteria

    b. Access to plant after accident

c. Difficulty of restoring reactor to service

Certain failure modes of a FVCS or a SCS could even aggravate accident consequences. As mentioned in UCS/NYPIRG's testimony, the probability of basemat penetration could be increased. Several other failure modes are listed below. The first two of these failure modes are listed in NUREG/CR-1410, a document referenced by UCS/NYPIRG.

1. Subatmospheric Pressures in Containment Building

Should an accident occur at one of the Indian Point plants, a mixture of steam, air, and entrained water droplets would be created in the present containment. This mixture, because of its higher pressure, would either pass through the FVCS or expand into the SCS upon opening of an isolation valve. In either case, the original containment air would be pushed out of the containment leaving a mostly steam atmosphere. Should the containment, now mostly filled with steam, go through a depressurization because of initiation of the sprays or fans, or just natural condensation cooling, the containment would likely become subatmospheric (less than 15 psia). In general, containments have far greater superatmospheric capabilities than subatmospheric capabilities. Such an event could then lead to early containment failure and early fatalities.

2. Hydrogen Explosions

UCS/NYPIRG expresses a concern about the effects of hydrogen explosions on containment integrity. The analysis in IPPSS shows that this is not a significant concern for the Indian Point containments. Yet the very designs UCS/NYPIRG suggests promote explosive hydrogen mixtures within the FVCS or SCS. It is well established that hydrogen-air mixtures are more explosive than hydrogen-air-steam mixtures. Both the FVCS and the SCS could remove the steam from the containment into the vent system, thereby possibly promoting explosive mixtures within the system.

3. Differential Motion

Should a seismic event occur, differential motion between the present containment and a FVCS or SCS, which would be located some distance away due to site space availability, might cause failure of any connecting structure that links the containment to the proposed modification. Such a failure could lead to early fatalities.

4. Isolation Difficulties

Valves that isolate the FVCS and SCS from the containment would be subject to a number of failure modes. These valves may fail to open, negating the use of the FVCS and SCS. They might be opened prematurely or stick open, thereby placing an unacceptable load on the filters or suppression pool. Some of these failures could be initiated by equipment problems or operator errors.

In summary, the immature state of FVCS and SCS technology suggests that there may be numerous unidentified failure modes. Although some tentative findings as to the potential impact of a filtered source term are presented in the IPPSS, these results reflect a partial preliminary effort without any detailed evaluation of the factors set forth above at pages 14 to 17. UCS/NYPIRG's testimony fails to offer an Indian Point-specific design for a filtered vent. Accordingly, the IPPSS preliminary effort offers little insight into the value of the hypothetical filtered vent proposed by UCS/NYPIRG.

No comprehensive risk analysis has been made of any of these modifications joined to and interacting with the present Indian Point containments. Consequently, their risk reduction worth has not adequately been established. Further, no regulatory guidance has been issued with regard to the design, licensing, operation, or testing of the FVCS and SCS. With an adequate research and development program, an actual design, and an associated probabilistic risk assessment, the frequency of these potential failure modes of the FVCS and SCS can be minimized. However, because of the immaturity of these technologies, such a process would be long and costly.

F. Source Term Sensitivity Analysis

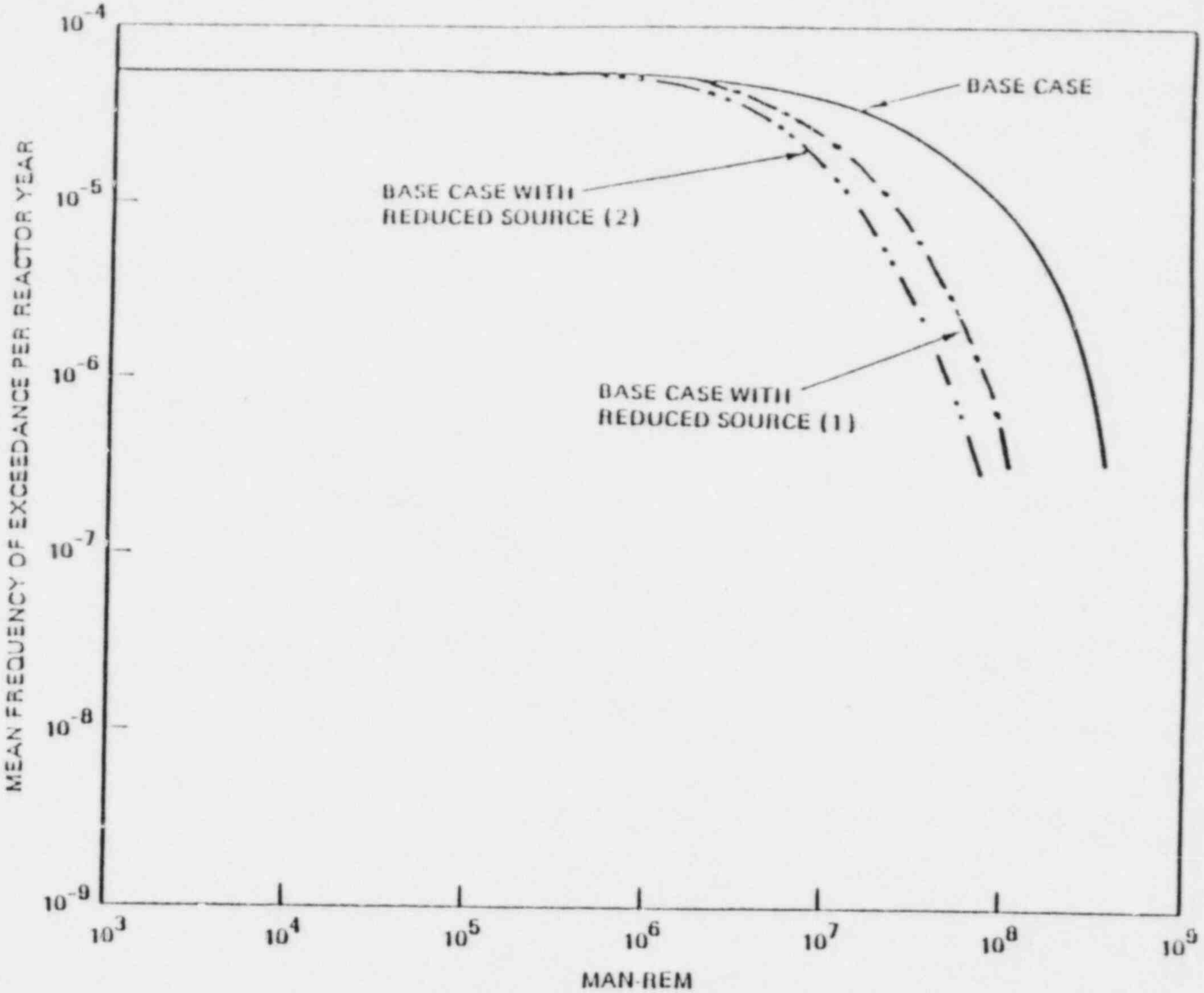
As an outgrowth of the Three Mile Island accident, there is keen interest and a tremendous amount of ongoing work in private industry, government regulatory agencies, and various national laboratories regarding source terms, that is, the amount and mix of radionuclides that would be released under various postulated accident scenarios. Releases of radionuclides smaller than those assumed in the IPPSS would result in a significant reduction in the risk reported in the IPPSS, as well as a significant diminution of the value of all mitigating features. Testimony justifying the use of smaller sources will be presented elsewhere in this hearing under Question 1 for the scenarios that dominate the Indian Point risk. For the purposes of evaluating the potential for risk reduction from a FVCS or a SCS, it is useful to present a source term sensitivity analysis.

Figures 1 and 2 show the sensitivity of the whole body man-rem dose to source term reduction for Indian Point 2 and 3, respectively. Three curves can be compared: the IPPSS man-rem risk curve, a curve where the source term assumed in the IPPSS is reduced by a factor of 10, and a curve where the reduction factor is 20. Only the iodine and particulates are reduced in these latter curves; noble gas releases are unaffected.

Figures 1 and 2 show significant reduction in the latent effects with source term reductions of 10 and 20. For example, a 20-fold reduction in the source term results in almost 10-fold reduction in the maximum man-rems. Figures 3 and 4 show similar reductions for the latent fatality risk. Even if the IPPSS source term is used, both Indian Point units are well within the Commission safety goal limits for latent fatality risk. Smaller source terms will increase this margin further.

The Commission has recognized the great importance of updating its understanding of source term technology and has a large research program well underway in the area, as does the industry in its IDCOR program. Interim results are expected in the near future. The benefits of mitigating devices such as FVCS and SCS are measured in terms of man-rems averted. These benefits decrease with smaller source terms. Because of the lower risk associated with anticipated smaller source terms, the Commission has deferred major policy decisions so as to base them on this newer information. This same reasoning should also be applied to any proposal for a FVCS and SCS, the precise worth of which can only be evaluated after source terms appropriate for the purpose have been determined.

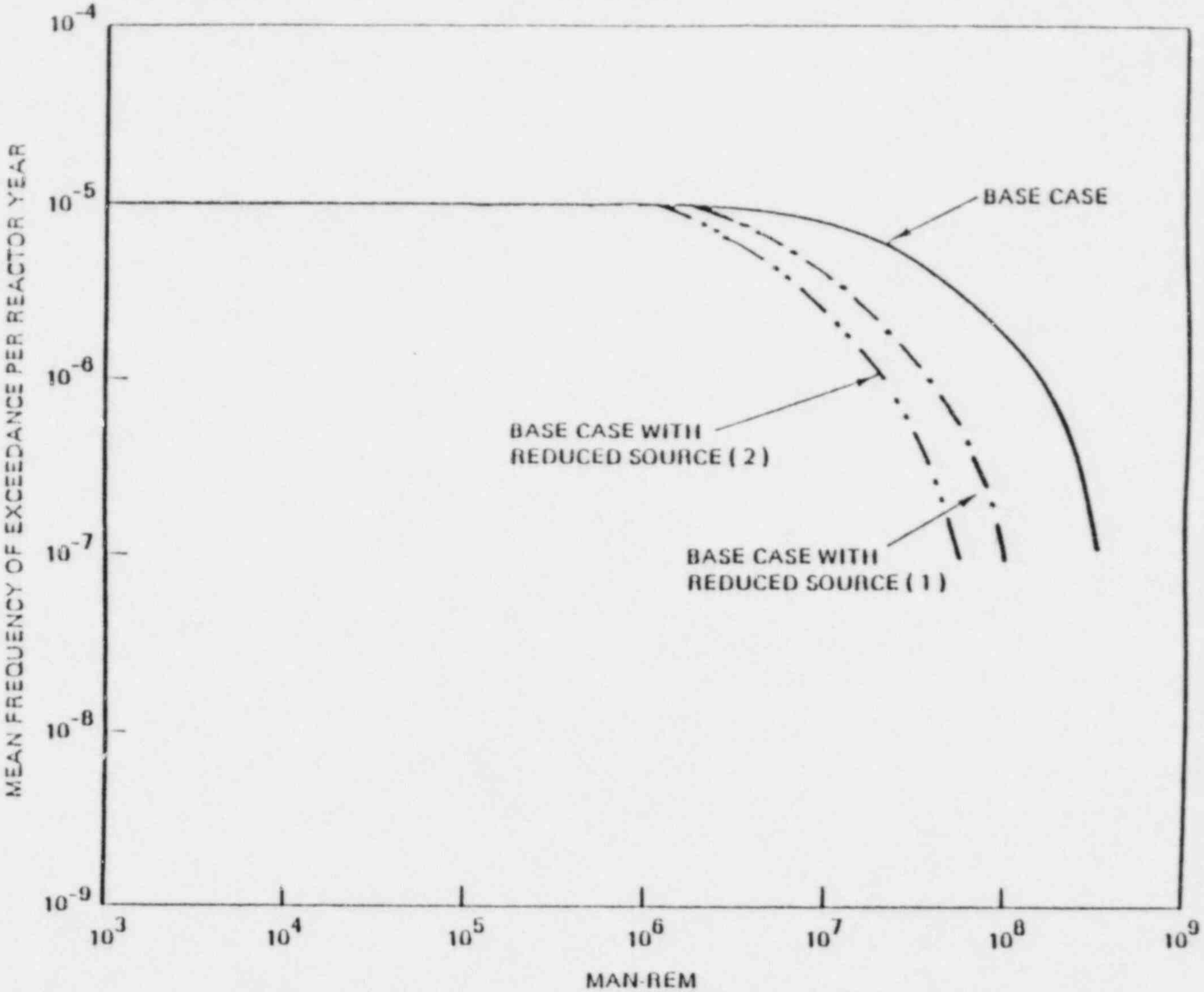
FIGURE 1



EFFECT OF REDUCED SOURCE TERM ON WHOLE BODY MAN-REM - INDIAN POINT 2

Curve (1): Iodine and particulates reduced by a factor of 10.
Curve (2): Iodine and particulates reduced by a factor of 20.

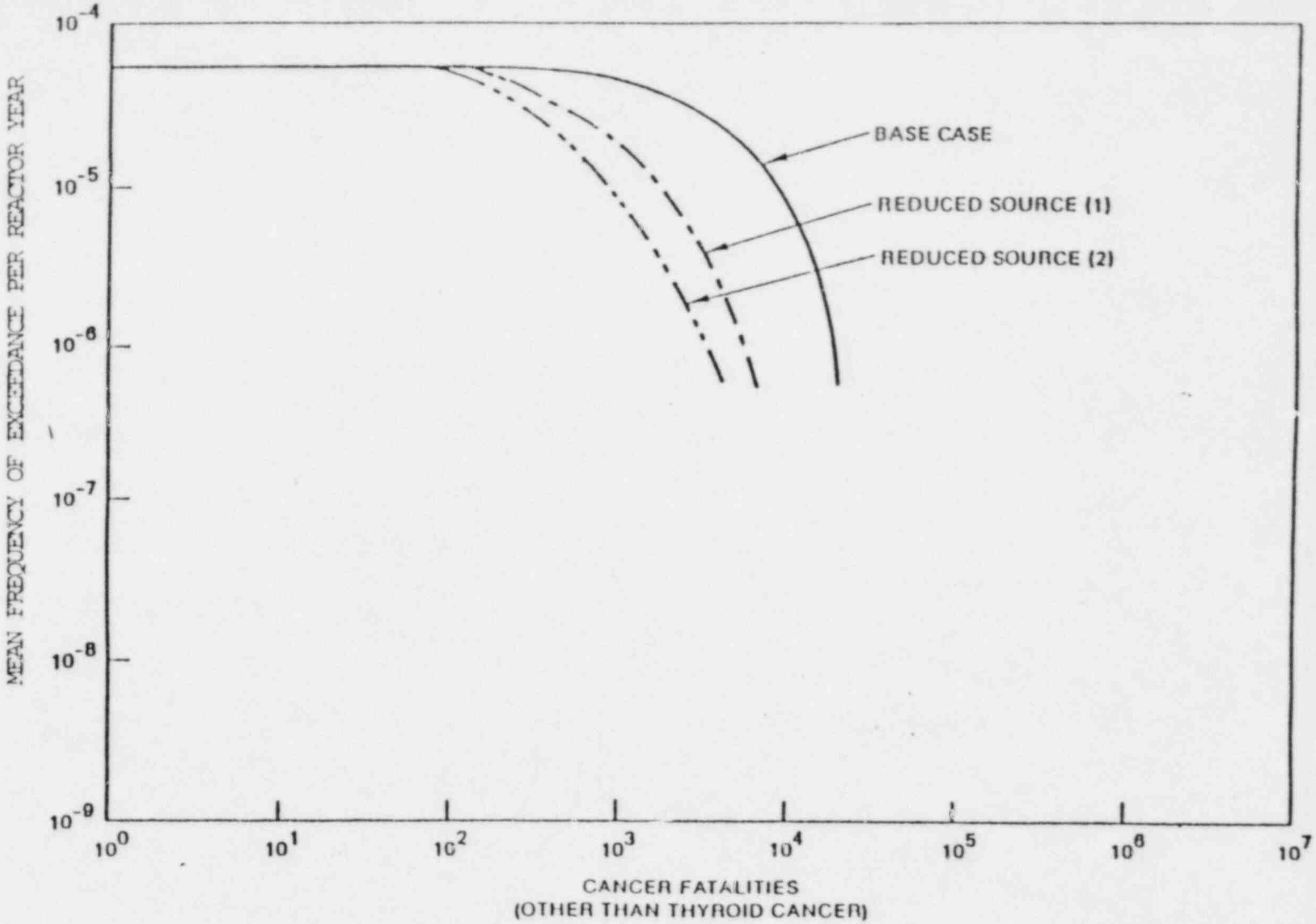
FIGURE 2



EFFECT OF REDUCED SOURCE TERM ON WHOLE BODY MAN-REM - INDIAN POINT 3

Curve (1): Iodine and particulates reduced by a factor of 10.
Curve (2): Iodine and particulates reduced by a factor of 20.

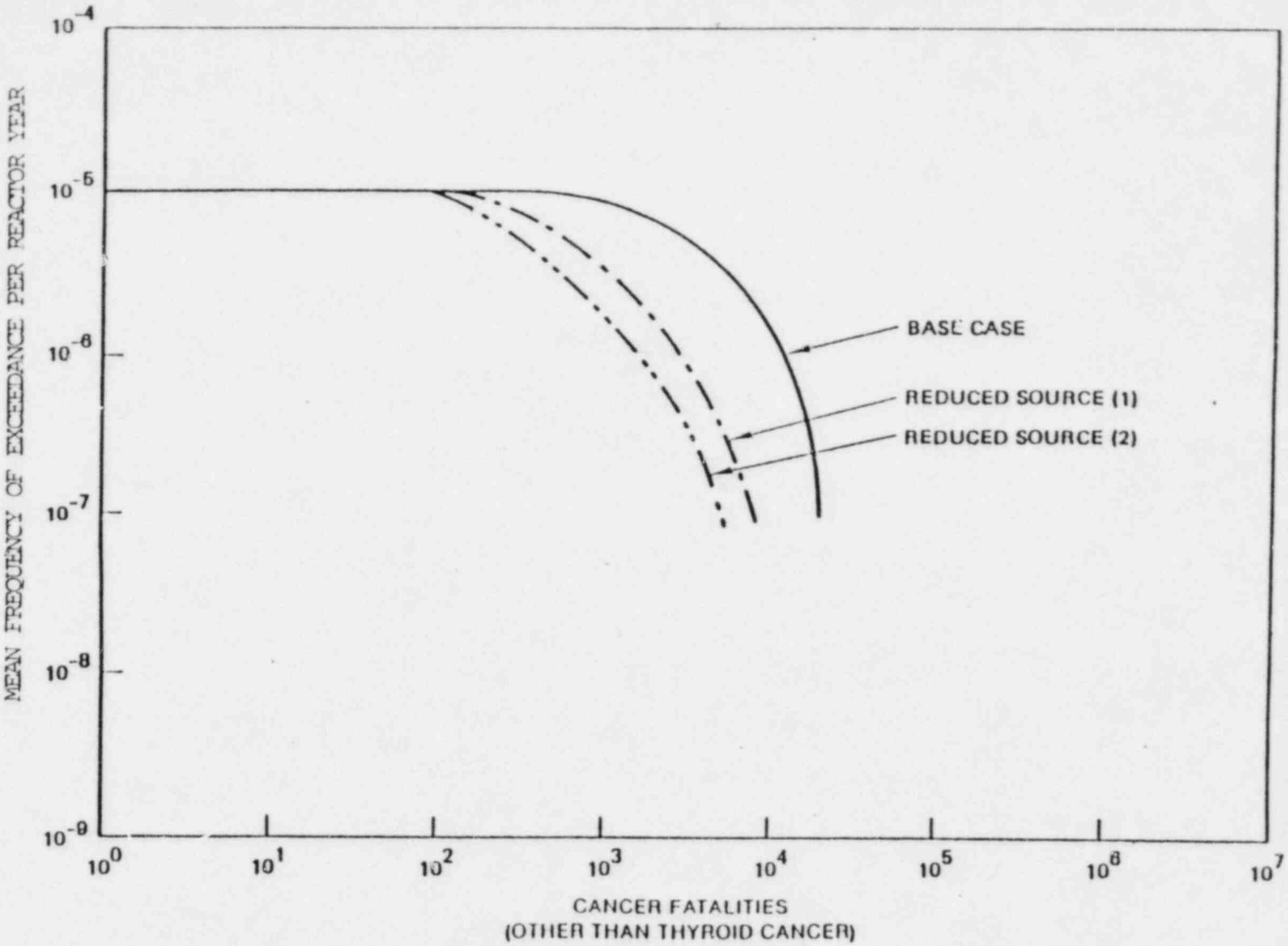
FIGURE 3



EFFECT OF REDUCED SOURCE TERM ON LATENT CANCER FATALITIES - INDIAN POINT 2

Curve (1): Iodine and particulates reduced by a factor of 10.
Curve (2): Iodine and particulates reduced by a factor of 20.

FIGURE 4



EFFECT OF REDUCED SOURCE TERM ON LATENT CANCER FATALITIES - INDIAN POINT 3

Curve (1): Iodine and particulates reduced by a factor of 10.
Curve (2): Iodine and particulates reduced by a factor of 20.

G. Conclusions

The installation of a filtered vented containment system or a separate containment system at Indian Point 2 and 3 is not justified for the following reasons:

- The risk from operation of these plants, as determined in the IPPSS, is very low.
- Such systems are only potentially useful in reducing latent fatality risks, yet both Indian Point 2 and Indian Point 3 meet the Commission safety goal for this health effect.
- The pressure boundary of the present Indian Point containment should not be changed and possibly compromised.
- FVCS and SCS would not reduce the frequency or types of accident initiators.
- FVCS and SCS would not reduce the frequency of core melts.
- FVCS and SCS would not reduce the early fatality risk.
- FVCS and SCS technology is immature and unproven.
- The possibility of a negative impact of these features has not been considered by UCS/NYPIRG.
- The worth of any mitigating feature is highly dependent on the source term. Consistent with previous Commission decisions, no decision to install a device should be made until the research program is concluded.
- The IPPSS identified risk contributors amenable to improvement by plant modifications which are being implemented by the licensees and yield substantial reductions in risk.

Dennis C. Richardson - Risk Assessment Technology Manager

Penn State University, B.S. Aerospace Engineering

1963

M.S. Control Engineering

1965

San Diego State University, M.S. Mathematics

1970

University of Pittsburgh, MBA

1980

Mr. Richardson has many years of professional and management experience in the nuclear field. He joined the Pressurized Water Reactor Division of Westinghouse in 1972 where he managed the Reactor Protection Analysis Group for performing nuclear plant safety analysis and, most recently, has managed the Risk Assessment Technology Organization.

Prior to this, Mr. Richardson was with Gulf General Atomic where he worked on design of control and safety systems for the gas-cooled nuclear plant. At Westinghouse, he has participated in and directed a number of risk assessment and safety analysis studies for a wide variety of applications. He was a principal investigator in both the Zion Station and Indian Point Station Reactor Safety Studies. He directed the PRA studies for the Westinghouse Owners Group that addressed the Post-TMI NUREG requirements on emergency procedures and operator display requirements. Mr. Richardson was technical and program manager for the British (NNC) Reference Water Reactor Safety Study. He has also led the development of economic and financial risk assessment techniques for the use in new reactor model design concepts.

Mr. Richardson is a member of the IEEE and ANS and has served on the working groups for two standards committees. He is reviewing the sections for the PRA manual directed by NRC to be finished in 1981. He is author or co-author of more than 15 reports and papers dealing with risk assessment and various aspects of nuclear plant design

* * * * * *

NAME

DENNIS C. BLEY

EDUCATION

Ph.D., Nuclear Reactor Engineering, Massachusetts Institute of Technology, 1979.
Courses in nuclear engineering and computer science, Cornell University, 1972-1974.
U.S. Navy Nuclear Power School, 1968.
University of Cincinnati, B.S.E.E., 1967.
Courses in Mathematics and Physics, Centre College of Kentucky, 1961-1963.

PROFESSIONAL EXPERIENCE

General Summary

A consultant at Pickard, Lowe & Garrick, Inc., 1979-present. Technical analysis of power plant availability and risk. Cost-benefit analysis of power plant system changes. Preparation of technical reports, expert testimony, and proposals. Supervision of the technical quality of PLG reports and direction of some PLG projects. Instructor at availability, risk, and decision analysis courses offered by PLG. Oyster Creek Probabilistic Risk Assessment (OPSA). Assisted in the completion and review of this complete risk assessment of an operating BWR performed for Jersey Central Power & Light. Work Order Scheduling System (WOSS). Assisted in developing the San Onofre 2 and 3 plant model for a computer based work order prioritizing, scheduling, and record keeping system for Southern California Edision Company. Steam Turbine Diagnostics Cost-Benefit Analysis. Developed and applied a procedure for evaluating diagnostic alternatives for EPRI. Reliability Analysis of Diablo Canyon Auxiliary Feedwater System for Pacific Gas & Electric. Midland Plant Auxiliary Feedwater System Reliability Analysis for Consumers Power. Technical Review of the "Office of Emergency Services Recommended Emergency Planning Zone Considerations..." for Southern California Edison. Prioritization of NRC Action Plan for NSAC. Development of a methodology and participation in an AIF workshop to apply it for EPRI/NSAC. Zion and Indian Point Probabilistic Safety Studies. Methods development, systems analysis, and plant modeling. Other PRAs--LaSalle, Browns Ferry, Midland, Pilgrim 1, and Oconee.

On USS Enterprise, Reactor Training Assistant, 5 months, 1971. Responsible for technical training of approximately 400 nuclear trained officers and men prior to annual safeguards examination. Propulsion Plant Station Officer, 9 months, 1970-1971. Responsible for maintenance and operation of one propulsion plant (two reactors, eight steam generators, and associated equipment) during power range testing of new reactors and during deployment. Approximately 50 enlisted personnel were assigned to the plant. Shift Propulsion Plant Watch Officer, 15 months, 1969-1970. Supervised a crew of about 20 navy enlisted operators and many shipyard workers on 8-hour shift rotation conducting maintenance and testing in one propulsion plant during refueling-overhaul. Shipboard qualifications: Propulsion Duty Officer, responsible for all propulsion equipment during absence of Reactor Officer and Engineer Officer. Engineering Officer of the Watch, operational watch in Central Control, responsible for all propulsion and engineering equipment and watch standers. Propulsion Plant Watch Officer, operational watch in one propulsion plant, directed and responsible for all operations in the plant.

At Cincinnati Bell, Plant staff assistant, 4 months, 1967. Worked in central office and transmission group supplying technical assistance to the line organization. Cooperative trainee, 3 years, 1964-1967, work-study program with alternate three month periods at the University of Cincinnati.

Chronological Summary

1979-Present	Consultant, Pickard, Lowe and Garrick, Inc.
1974-1979	Massachusetts Institute of Technology. Research assistant for Department of Energy LWR Assessment Project. Teaching assistant in engineering of nuclear reactors.
Summer 1976	Northeast Utilities. Engineer: economy studies, plant startup, analysis of physics tests.
1967-1974	U.S. Naval Reserve, active duty. Instructor of naval science, Cornell University, 1971-1974; Reactor Department of USS Enterprise, deployment and refueling-overhaul, 1969-1971; Nuclear Power training program and Officer Candidate School, 1967-1969.
1964-1967	Cincinnati Bell. Plant staff assistant and work-study program trainee.

MEMBERSHIPS, LICENSES, AND HONORS

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American Nuclear Society.
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Sherman R. Knapp Fellowship (Northeast Utilities), 1975-1976.
Sloan Research Traineeship, 1974-1975.
Eta Kappa Nu (national electrical engineering honors society), 1967.

REPORTS AND PUBLICATIONS

"Seabrook Probabilistic Safety Assessment," Public Service Company of New Hampshire, to be published in 1983.

Pickard, Lowe and Garrick, Inc., "Midland Probabilistic Risk Assessment," Consumers Power Company, to be published in 1982.

Oconee Probabilistic Risk Assessment," a joint effort of the Nuclear Safety Analysis Center, Duke Power, and other participating utilities, to be publisned in 1982.

Tennessee Valley Authority and Pickard, Lowe and Garrick, Inc., "Browns Ferry Probabilistic Risk Assessment," to be published in 1982.

Apostolakis, G., M. Kazarians, and D. C. Bley, "A Methodology for Assessing the Risk from Cable Fires," accepted for publication in Nuclear Safety, 1982.

Kaplan, S., H. F. Perla, and D. C. Bley, "A Methodology for Seismic Safety Analysis of Nuclear Power Plants," proposed presentation at the International Meeting on Thermal Nuclear Reactor Safety, Chicago, Illinois, August 29-September 2, 1982.

Bley, D. C., S. Kaplan, and B. J. Garrick, "Assembling and Decomposing PRA Results: A Matrix Formalism," proposed presentation at the International Meeting on Thermal Nuclear Reactor Safety, Chicago, Illinois, August 29-September 2, 1982.

Garrick, B. J., S. Kaplan, and D. C. Bley, "Recent Advances in Probabilistic Risk Assessment," prepared for the MIL Nuclear Power Reactor Safety Course, Cambridge, Massachusetts, July 19, 1982.

Fleming, K. N., S. Kaplan, and B. J. Garrick, "Seabrook Probabilistic Safety Assessment Management Plan,"PLG-0239, June 1982.

Garrick, B. J., "Lessons Learned From First Generation Nuclear Plant Probabilistic Risk Assessments," to be presented at the Workshop on Low-Probability/High-Consequence Risk Analysis, Arlington, Virginia, June 15-17, 1982.

Garrick, B. J., S. Kaplan, D. C. Iden, E. B. Cleveland, H. F. Perla, D. C. Bley, D. W. Stillwell, H. V. Schneider, and G. Apostolakis, "Power Plant Availability Engineering: Methods of Analysis, Program Planning, and Applications," EPRI NP-2168, PLG-0165, May 1982.

Bley, D. C., and R. J. Mulvihill, "Comments on Evaluation of Availability Improvement Options for Moss Landing Units 6 and 7," PLG-0226, March 1982.

Stillwell, D. W., G. Apostolakis, D. C. Bley, P. H. Raabe, R. J. Mulvihill, S. Kaplan, and B. J. Garrick, "EEI Availability Handbook," PLG-0218, January 1982.

Bley, D. C., L. G. H. Sarmanian, and D. W. Stillwell, "Reliability Analysis of Safety Injection System Modification, San Onofre Nuclear Generating Station - Unit 1," PLG-0206, October 1981.

"Zion Probabilistic Safety Study," Commonwealth Edison Company, September 1981.

Buttemer, D. R., "Analysis of Postulated Accidents During Low Power Testing at the San Onofre Nuclear Generating Station--Unit 2," PLG-0199, September 1981.

Bley, D. C., D. W. Stillwell, and R. R. Fray, "Reliability Analysis of Diablo Canyon Auxiliary Feedwater System," presented at the Tenth Biennial Topical Conference on Reactor Operating Experience, Cleveland, Ohio, August 17-19, 1981.

Garrick, B. J., and D. C. Bley, "Lessons Learned from Current PRAs," presented to the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment, Los Angeles, California, July 28, 1981.

Kaplan, S., G. Apostolakis, B. J. Garrick, D. C. Bley, and K. Woodard, "Methodology for Probabilistic Risk Assessment of Nuclear Power Plants," draft version of a book in preparation, PLG-0209, June 1981.

Perla, H. F., "Project Plan: Probabilistic Risk Assessment, Midland Nuclear Power Plant," PLG-0150, May 1981.

Bley, D. C., C. L. Cate, D. W. Stillwell, and B. J. Garrick, "Midland Plant Auxiliary Feedwater System Reliability Analysis Synopsis," PLG-0166, March 1981.

Pickard, Lowe and Garrick, Inc., "A Methodology to Quantify Uncertainty of Cost of Electricity for Alternate Designs of (Combustion) Turbine Combined Cycle Plants," PLG-0162, March 1981.

Garrick, B.. J., S. Ahmed, and D. C. Bley, "A Methodology for Evaluating the Costs and Benefits of Power Plant Diagnostic Techniques," submitted for presentation at the Ninth Turbomachinery Symposium, Houston, Texas, December 9-11, 1980.

Pickard, Lowe and Garrick, Inc., "Seminar: Probabilistic Risk Assessment of Nuclear Power Plants," PLG-0154, November 1980.

Pickard, Lowe and Garrick, Inc., "Project Plan: Probabilistic Risk Assessment, Browns Ferry Nuclear Plant Unit 1," PLG-0149, October 1980.

Garrick, B. J., S. Kaplan, D. C. Iden, E. B. Cleveland, H. F. Perla, D. C. Bley, and D. W. Stillwell, "Power Plant Availability Engineering, Methods of Analysis - Program Planning - Applications," 2 Vols., PLG-0148. October 1980.

Bley, D. C., C. L. Cate, D. W. Stillwell, and B. J. Garrick, "Midland Plant Auxiliary Feedwater System Reliability Analysis," PLG-0147, October 1980.

Bley, D. C., D. M. Wheeler, C. L. Cate, D. W. Stillwell, and B. J. Garrick, "Reliability Analysis of Diablo Canyon Auxiliary Feedwater System," PLG-0140, September 1980.

Garrick, B. J., et al, "Project Plan: Oconee Probabilistic Risk Assessment," PLG-0138, August 1980.

Garrick, B. J., D. M. Wheeler, E. B. Cleveland, D. C. Bley, L. H. Reichers, and C. B. Morrison, "Operating Experience of Large U.S. Steam Turbine-Generators; Volume 1 - Data, Volume 2 - Utility Directory," PLG-0134, June 1980.

Garrick, B. J., S. Kaplan, and D. C. Bley, "Seminar: Power Plant Probabilistic Risk Assessment and Reliability," PLG-0127, May 1980.

Garrick, B. J., and S. Kaplan, "Oyster Creek Probabilistic Safety Analysis (OPSA)," presented at the ANS-ENS Topical Meeting on Thermal Reactor Safety, Knoxville, Tennessee, April 8-11, 1980.

Garrick, B. J., S. Kaplan, G. E. Apostolakis, D. C. Bley, and T. E. Potter, "Seminar: Probabilistic Risk Assessment as Applied to Nuclear Power Plants," PLG-0124, March 1980.

Garrick, B. J., S. Ahmed, and D. C. Bley, "A Methodology for Evaluating the Costs and Benefits of Power Plant Diagnostic Techniques," PLG-0118, January 1980.

Kaplan, S., B. J. Garrick, and D. C. Bley, "Notes on Risk, Probability, and Decision," PLG-0113, November 1979.

Bley, D. C., C. L. Cate, D. C. Iden, B. J. Garrick, and J. M. Hudson, "Seismic Safety Margins Research Program (Phase I), Project VII - Systems Analysis," PLG-0110, September 1979.

Cate, C. L., and B. J. Garrick, "W-501 Combustion Turbine Starting Reliability Analysis," PLG-0103, June 1979.

Pickard, Lowe and Garrick, Inc., "Plant Availability Program Specification, San Onofre Nuclear Generating Station," March 1979.

Pickard, Lowe and Garrick, Inc., "Work Order Scheduling System, Design Specification," March 1979.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:
James P. Gleason, Chairman
Frederick J. Shon
Dr. Oscar H. Paris

In the Matter of	
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point, Unit No. 2)	Docket Nos. 50-247 SP 50-286 SP
POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	Jan. 17, 1983

CERTIFICATE OF SERVICE

I hereby certify that on the 17th day of January, 1983, I caused a copy of Licensees' Testimony of Dennis C. Bley and Dennis C. Richardson on Contentions 2.1(a) and 2.1(d) to be hand delivered to those parties marked with an asterisk, and served by first class mail, postage prepaid on all others.

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