UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CONSOLIDATED EDISON COMPANY OF NEW YORK	)	Docket Nos. 50-247-SP
(Indian Point Unit 2)	)	50-286-SP
	)	
POWER AUTHORITY OF THE STATE OF NEW YORK	)	28 December 1982
(Indian Point Unit 3)	)	

UCS/NYPIRG TESTIMONY OF
GORDON R. THOMPSON AND STEVEN C. SHOLLY
ON COMMISSION QUESTION TWO,
CONTENTIONS 2.1(a) AND 2.1(d)

Q.01 Would you please identify yourselves?

A.01 WITNESS THOMPSON: My name is Gordon R. Thompson, and I am a Senior Staff Scientist with the Union of Concerned Scientists in Cambridge, Massachusetts.

WITNESS SHOLLY: My name is Steven C. Sholly, and I am a Technical Research Associate with the Union of Concerned Scientists in Washington, D.C.

Q.02 Have you prepared statements of professional qualifications?

A.02 Yes. The statements of professional qualifications are attached to this testimony.

Q.03 To which Commission Question in this special investigation is this testimony addressed?

A.03 This testimony addresses Commission Question Two which states:

DS03

What improvements in the level of safety will result from required or referenced in the Director's Order to the Licensee, dated February 11, 1980? (A contention by a party that one or more specific safety measures in addition to those identified or referenced by the Director, should be required as a condition of operation would be within the scope of this inquiry if, according to the Licensing Board, admission of the contentions seems likely to be important to resolving whether: (a) there exists a significant risk to public health and safety, notwithstanding the Director's measures, and (b) the additional proposed measures would result in a significant reduction in that risk.)

Q.04 To which Contentions is this testimony addressed?

A.04 This testimony addresses Board Contentions 2.1(a) and 2.1(d) which state:

CONTENTION 2.1(a)

A filtered vented containment system for each unit must be installed.

CONTENTION 2.1(d)

A separate containment structure must be provided into which excess pressure from accidents and transients can be relieved without necessitating releases to the environment, thereby reducing the risk of containment failure by overpressurization.

Q.05 What is the nature of the risk to the public posed by operation of Indian Point Units 2 and 3?

A.05 The risk[1] to the public from the operation of Indian Point Units 2 and 3 arises from the potential for releases of radioactive material from the facilities to the environment.

Q.06 Under what conditions could such releases occur?

A.06 Releases of radioactive materials to the environment from Indian Point Units 2 and 3 can occur due to a variety of causes associated with normal operations. In addition, releases of radioactive materials to the environment could occur as a result of a variety of accidents, including those leading to severe core damage and core melt.

Q.07 What type of event which leads to the release of radioactive materials to the environment poses the greatest risk to the public?

A.07 It is widely recognized that the risk to the public from operation of nuclear power plants is dominated by core melt accidents.

Q.08 What evidence exists to support this conclusion?

A.08 The Reactor Safety Study (WASH-1400, NUREG-75/014, N.C. Rasmussen, et. al., Reactor Safety Study: An Assessment of Accident Risks in U.S. Nuclear Power Plants, U.S. Nuclear Regulatory Commission, October 1975) provides the original basis for this conclusion, and this work has been confirmed by subsequent risk studies performed under the "Reactor Safety Study Methodology Applications Program" (RSSMAP) and the "Interim Reliability Evaluation Program" (IREP). WASH-1400 used probabilistic risk assessment (PRA) methodology to evaluate two reactors in detail (i.e., the Surry PWR and the Peach Bottom BWR) and then generalized the results to include the first 100 nuclear power plants to be licensed in the U.S. (including Indian Point Units 2 and 3).

The RSSMAP and IREP reports also used PRA methodology to evaluate core melt probability, but did so for a greater variety of reactors. Inasmuch as the core melt probabilities calculated in the RSSMAP and IREP studies are all greater than the Peach Bottom BWR evaluated in WASH-1400 (which formed part of the basis for the conclusion that core melt accident cominate risk to the public), these studies tend to confirm the WASH-1400 conclusion that core melt accidents dominate risk to the public.[2]

Further, a report by Brookhaven National Laboratory compared the relative risks posed by normal operational releases, accidents within the design basis, and accidents exceeding the design basis (i.e., core melt accidents) for the Surry PWR. The report (NUREG/CR-0603, BNL-NUREG-90950, R.E. Hall, et. al., <u>A Risk Assessment of a Pressurized Water Reactor for Class 3-8 Accidents</u>, Brookhaven National Laboratory for the U.S. Nuclear Regulatory Commission, October 1979), expressing relative risk in terms of the curie equivalent Iodine-131 released per reactor-year, showed clearly that core melt accidents dominate public risk.

Finally, the Licensees' own <u>Indian Point Probabilistic Safety Study</u> (IPPSS) supports the general conclusion with specificity to Indian Point Units 2 and 3. Specifically, each of the release categories determined by IPPSS to contribute significantly to risk is associated with a core melt accident. In particular, WASH-1400, the RSSMAP and IREP reports, and IPPSS show that the largest releases of radioactive material occur for core melt accidents which result in containment failure with a release of radioactive material to the atmosphere.

Q.09 By what means can the risk from core melt accidents at Indian Point Units 2 and 3 be reduced?

A.09 In general, risk can be reduced by reducing accident probabilities, accident consequences, or both probability and consequences. Under present conditions, however, reduction of accident consequences appears to offer a greater opportunity for risk reduction than does reduction of accident probabilities.

Q.10 Why do you believe that measures to mitigate accident consequences are needed at Indian Point Units 2 and 3?

A.10 It has been noted (R.S. Denning and P. Cybulskis, "Reduction in Reactor Risk by Mitigation of Accident Consequences", Nuclear Safety, Vol. 22, No. 2, March-April 1981, pages 165-172) that once the few accident sequences which typically dominate risk have been identified and corrective measures have been implemented, the prospects for further reduction in risk by addressing probability are limited. This arises because there are typically a large number of accident sequences requiring preventive measures after the few dominant sequences have been addressed.

Despite the considerable efforts taken in design, fabrication, construction, operation, maintenance, and regulation of nuclear power plants, core melt accidents dominate risk. Much of the safety effort to date appears to have been driven by probability considerations. In essence, therefore, that effort has amounted to (at least with respect to core melt accidents) an accident prevention program. Despite this effort, however, a number of studies exist which together indicate that accidents involving severe core damage and core melt have a relatively high probability.

The recent "Accident Precursor Study" (NUREG/CR-2497, ORNL/NSIC-182, J.W. Minarick and C.A. Kukielka, Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report, Science Applications, Inc., Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission, June 1982) concluded, based on an examination of the

observed frequency of precursor events between 1969 and 1979, that the probability of severe core damage was between 1:222 and 1:588 per reactor-year.

In addition, for the nine reactors thus far evaluated in NRC program using PRA methodology, calculated core melt probabilities ranged from 1:500 to 1:34,500 per reactor-year.[3] It should be noted that these results contain large uncertainties.

Significant uncertainties have been identified for Indian Point Units 2 and 3 specifically. For the Indian Point reactors, the IPPSS report calculated a mean core melt probability from all causes of 1:2,500 per reactor-year for Indian Point Unit 2 and 1:11,000 per reactor-year for Indian Point Unit 3. However, the draft Sandia review of IPPSS[4] recalculated the probabilities for several release categories discussed in IPPSS, and found that for on release category alone5 the recalculated probabilities were 1:770 per reactor-year for Indian Point Unit 2 and 1:4,170 per reactor-year for Indian Point Unit 3.

Q.11 What evidence do you have that efforts to reduce risk by addressing only accident probabilities will not afford sufficient risk reduction to render mitigative measures unnecessary?

A.11 The difficulties of reducing risk by addressing accident probabilities are exemplified by the actions taken by the Licensees in response to the February 11, 1980, "confirmatory orders" issued by the Director of Nuclear Reactor Regulation of the NRC. Despite the number and variety of the requirements imposed[6] (many of which were aimed at reducing the probability of various accidents at Indian Point Units 2 and 3), the NRC

Staff calculated that the risk reduction afforded by these measures was a factor of about three. The NRC Staff itself has questioned the statistical significance of this conclusion (SECY-80-283, NUREG-0715, R.M. Bernero, et. al., Report of the Task Force on Interim Operation of Indian Point, U.S. Nuclear Regulatory Commission, 12 June 1980, page 26).

Q.12 Where should efforts be placed on engineered design features to have the greatest impact on reducing reactor accident consequences?

A.12 Such features should be capable of reducing the probability of containment failure and/or reducing the consequences of such failures. There are many factors which significantly impact the magnitude of accident consequences. However, given that we have no control over meteorology, a major factor over which we can exert some control in determining the magnitude of reactor accident consequences is the mode of containment failure.

Q.13 What mechanisms have been proposed by which containment failure might occur for Indian Point Units 2 and 3?

A.13 A number of studies have evaluated containment failure modes for Indian Point Units 2 and 3. These studies have identified the following containment failure modes for Indian Point Units 2 and 3:[7]

a. Steam explosion in-vessel leading to missile penetration of the containment;

b. Overpressurization by steam and non-condensibles;

c. Hydrogen burning;

d. Containment isolation failure;

e. Containment bypass due to interfacing systems LOCA;

f. Containment bypass due to massive steam generator tube rupture caused by in-vessel steam spike;

g. Penetration of containment basemat by molten core materials; and

h. Seismic structural failure of containment before core melt.

Q.14 Which of these failure modes have been found to be dominant?

A.14 Different studies have reached varying conclusions regarding which containment failure mode is dominant for Indian Point Units 2 and 3. NUREG-0850, Vol. 1, concluded that hydrogen burn is the dominant containment failure mode for Indian Point Units 2 and 3, and that elimination of overpressurization failures would reduce risk by only a factor of two. NUREG/CR-1410, on the other hand, concluded that (based on WASH-1400 data) overpressurization contributed about 90% to the total risk. IPPSS also identified overpressurization as the most frequent containment failure mode (associated with release category 2RW, referenced above in response to Q.10).

Q.15 How can containment failure be prevented or its consequences mitigated?

A.15 In general, the answer lies in increasing the capability of the containments at Indian Point Units 2 and 3 to withstand the stresses imposed by core melt accidents. Containment design is generally based on a "design basis accident", usually a loss of coolant accident (LOCA), and is not based on core melt accidents. Conservatism in containment design provides some margin above the design pressure for Indian Point Units 2 and 3 (62 psia according to NUREG-0850, Vol. 1, page 3-5).

The precise margin of safety provided by this design conservatism is a critical question. Various estimates of the failure pressure for the Indian Point Units 2 and 3 containments have ranged from 100 to 140 psia. These results, however, are based on analyses and assumptions rather than the results of scale model tests. A recent Sandia National

Laboratories report or containment safety margins (NUREG/CR-2549, SAND82-0324, Thomas E. Blejwas, et. al., Background Study and Preliminary Plans for A Program on the Safety Margins of Contaiments, Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, May 1982) raises issues affecting the uncertainty of calculation of containment failure pressures for Indian Point Units 2 and 3.

The study notes that the existing computer codes for structural analysis of reactor containments have not been qualified by comparison with tests-to-failure of containment-like structures (NUREG/CR-2549, page 9). The study concludes that only by obtaining experimental data can adequate judgments be made about how to apply these computer codes to actual containment structures. Regarding model tests which have been conducted, the Sandia study notes that a steel containment liner (required on U.S. concrete containments) was modeled in only one of the tests, and that none of the tests included validation of the codes across a range of model scales (NUREG/CR-2549, page 10). The report also noted that some of the tests which modeled large penetrations showed that these penetrations had a significant effect on containment response and initial leakage (Ibid.). Sandia concluded that if functional failure is to be simulated with scale models, the models should include containment liners, penetrations, and other "structural discontinuities" (NUREG/CR-2549, page 19).

It should be noted that even if the containment failure pressure for Indian Point Units 2 and 3 is as high as 140 psia as predicted in NUREG-0850, Vol. 1, and in IPPSS, the authors of these reports could not

rule out containment failure. Thus, it is necessary to examine the methods by which the existing Indian Point Units 2 and 3 containments could be improved to better withstand the stresses imposed by core melt accidents.

Q.16 What measures might be taken to improve containment capability for Indian Point Units 2 and 3?

A.16 A 1978 Sandia Laboratories study (NUREG/CR-0165, SAND77-1344, David D. Carlson and Jack W. Hickman, A Value-Impact Assessment of Alternate Containment Concepts, Sandia Laboratories for the U.S. Nuclear Regulatory Commission, June 1978) evaluated the following alternate containment concepts:

a. Increased containment design pressure through structural strengthening of traditional surface containments;

b. Increased containment design pressure by construction of containments underground in either shallow or deep burial;

c. Venting of containment gases through a filtered venting system;

d. Compartment venting (venting of containment gases to a separate, stand-by containment structure);

e. Increase in containment free volume;

f. Decrease in thickness of the containment basemat to assure early meltthrough of the basemat as a source of containment pressure relief;

g. Condensation of steam by use of an ice condenser;

h. Condensation of steam by use of a pressure suppression pool;

i. Evacuated containment (inhibits combustion of hydrogen by maintenance of a low oxygen inventory in containment); and

j. Double containment concepts.

In evaluating these concepts for possible backfit to Indian Point Units 2 and 3, we have concluded that some of these concepts are either not feasible or not suitable for Indian Point. Some (concepts a, f, and j) would be expensive in relation to the risk improvement to be gained, especially when compared with other means of accomplishing the same purpose. Others are only variants of one another for Indian Point since concepts g and h could not be backfit to the existing containments but would have to be contained in a separate structure such as in c or d. Concept i appears to be limited to affecting hydrogen burns. Based on these considerations and the conclusion of the Sandia value-impact analysis of the above concepts (that filtered venting and compartment venting offered the greatest potential for risk reduction for the least impact; NUREG/CR-0165, page 47), we have focused on concepts c and d above, which are represented as Contentions 2.1(a) and 2.1(d), respectively.

Q.17 Describe filtered venting.

A.17 Filtered venting is a concept whereby excess pressure from the containment is relieved through a variety of filtration and steam condensation systems with a subsequent release to the environment of the filtered containment gases (the filtered effluent would contain mostly noble gases). The risk reduction potential from this concept derives from its dual function of prevention of catastrophic containment failures and filtering of the effluent to limit the release of radioactive materials

Q.18 Describe compartment venting.

A.18 Compartment venting is a concept whereby excess pressure from the containment is relieved to a second containment structure for

condensation of steam, control of hydrogen gas, and containment of radioactive materials. Unlike filtered venting, compartment venting prevents catastrophic containment failures and prevents the release of radioactivity to the environment.

Q.19 What type of filtered venting system do you prefer for Indian Point Units 2 and 3?

A.19 We prefer a gravel filter concept similar to the one intended for installation at the Barseback nuclear plant in Sweden (K. Johansson, et. al., "Design Consideration for Implementing a Vent-Filter System at the Barseback Nuclear Power Plant", paper for presentation at the International Meeting on Thermal Reactor Safety, 29 August-2 September 1982, Chicago, IL, FILTRA Log No. 255) and consider that it could be implemented at Indian Point Units 2 and 3. This concept can be relatively passive in operation and can be inherently resistant to hydrogen explosions. We recommend that the operator be provided with control over the valve between the containment and the filtered venting system to provide flexibility in determining when to initiate venting based on actual conditions existing during an accident. An automatic initiation feature should be included in the design as well to ssure that filtered venting is initiated once the containment pressure reaches a pre-determined value. In addition, design studies should also be conducted for concepts involving a filtered venting system based on a pressure suppression pool as discussed in NUREG/CR-1410.

Q.20 What type of compartment venting system do you prefer for Indian Point Units 2 and 3?

A.20 We prefer a compartment venting system which incorporates a pressure suppression pool and hydrogen igniters, and consider that this concept

could be implemented at Indian Point Units 2 and 3. The design would provide for hydrogen control in the secondary containment, thus providing assurance that the compartment would not fail as a result of hydrogen burns. As with the filtered venting system, we recommend that the operator be provided with control over the valve between the containment and the compartment venting system. An automatic initiation feature should be implemented for the reason discussed above with respect to filtered venting.

Q.21 Is it necessary to have a separate system for each unit at Indian Point?

A.21 Yes. This is necessary due to the significant risk of accidents at the two reactors having a common origin.

Q.22 Which containment failure modes are affected by the filtered venting and compartment venting systems proposed by UCS/NYPIRG?

A.22 In NUREG/CR-0165, Sandia Laboratories evaluated the impact of filtered venting and compartment venting on containment failure modes a, b, c, d, and g as listed in response to Q.13 above. This study concluded that both filtered venting and compartment venting could eliminate containment failure modes b and c (overpressurization and hydrogen burn) unless active components fail. The study also concluded that failure modes a and d (steam explosion/missile failure and containment isolation failure) would be unaffected by filtered venting and compartment venting. Sandia also found that failure mode g (basemat penetration) would be increased in likelihood.

In a separate study, a team at Battelle Columbus Laboratories (NUREG/CR-0138, BMI-2002, P. Cybulskis, et. al., Effect of Containment Vennting on the Risk from LWR Meltdown Accidents, Battelle Columbus

Laboratories for the U.S. Nuclear Regulatory Commission, June 1978) concluded that for PWR accidents involving loss of containment heat removal, venting the containment "could significantly delay the start of core melting and thus offer the possibility to effect repairs and avoid core melting" (NUREG/CR-0138, page 35). Battelle also concluded, in contrast to the Sandia evaluation discussed above, that venting before core meltdown could provide some benefit in the steam explosion/missile penetration scenario even if the containment fails subsequent to venting initiation (NUREG/CR-0138, page 4). Battelle concluded that containment venting could also have a benefit in the event of containment isolation failure in minimizing the release of radioactive materials to the environment (NUREG/CR-0138, page 4), also in contrast to the Sandia assessment in NUREG/CR-0165.

NUREG/CR-1410 noted that failure modes such as those involving containment bypass (interfacing systems LOCA and steam generator tube rupture scenarios) establish a "risk floor" below which a venting system cannot reduce risk since it would not affect these containment failure modes (NUREG/CR-1410, page 1.24). Although this report also raises a question about the Sandia conclusion in NUREG/CR-0165 regarding the ability of filtered venting and compartment venting systems to completely eliminate hydrogen burns as a containment failure mode, NUREG/CR-1410 concluded that a reduction of hydrogen burning risks is offered by filtered venting because such a system would remove available oxygen from the containment through the vent opening and would increase the difference between the incident containment pressure and the containment failure pressure (NUREG/CR-1410, page 1.26). The report

also notes that for compartment venting, a significantly faster venting rate (compated to filtered venting) could be used (NUREG/CR-1410, page 1.6'), thus increasing the rate at which hydrogen could be removed from the primary containment.

Thus, given effective venting system operation, filtered venting and compartment venting systems will effectively eliminate overpressurization failure of the Indian Point Units 2 and 3 containments, and will offer substantial protection against a range of hydrogen burn events. In addition, the capability for some reduction of releases in the event of containment isolation failure is provided. Also, as noted in NUREG/CR-1029 (NUREG/CR-1029, SAND79-1088, Allan S. Benjamin, Program Plan for the Investigation of Vent-Filtered Containment Conceptual Designs for Light Water Reactors, Sandia Laboratories for the U.S. Nuclear Regulatory Commission, October 1979, pages 29-30), filtered venting systems provide for an improvement in sabotage protection by increasing the difficulty of achieving a large release of radioactive materials to the environment. This is because a saboteur would not only have to initiate a core melt accident but would also have to incapacitate the filtered venting system or prevent its use by force. We consider that similar considerations apply to the compartment venting system.

Q.23 Are there other nuclear reactor facilities which have filtered venting systems or for which such systems are planned for the future?

A.23 Yes. The Swedish government has required implementation of a filtered venting system for the Barseback nuclear power plant before 1 September 1986 (K. Johansson, et. al., "Design Considerations for Implementing a

Vent-Filter System at the Barseback Nuclear Power Plants", paper intended for presentation at the International Meeting on Thermal Nuclear Reactor Safety, 29 August-2 September 1982, Chicago, IL).

In addition, a filtered venting system has been proposed by the applicants for a construction permit for the Clinch River Breeder Reactor Plant; few details are available, however (NUREG-0139, Supplement No. 1, Vol. 2, Supplement to Final Environmental Statement related to construction and operation of Clinch River Breeder Reactor Plant, U.S. Nuclear Regulatory Commission, Appendix J, October 1982).

Q.24 How effective are filtered venting and compartment venting systems projected to be in reducing consequences of containment failure by overpressurization?

A.24 NUREG/CR-0165 approached this question by comparing WASH-1400 consequence results with the results that would be obtained by a variety of alternate containment concepts. In this evaluation, the "expected value" and the "maximum calculated value" of early fatalities and latent cancer fatalities were assigned a value of 1.00, and the relative risk for filtered venting and compartment venting were assessed. The study found that filtered venting and compartment venting would have the same impact (NUREG/CR-0165, page 40): "expected" early fatalities would be reduced to 0.08 and the "maximum calculated" early fatalities would be reduced to 0.29 relative to the existing plants; in addition, "expected" cancer deaths would be reduced to 0.11 and "maximum calculated" cancer deaths would be reduced to 0.29 relative to existing plants. It should be noted that filtered venting concept evaluated in NUREG/CR-0165 used a pressure suppression pool in a separate building, and the compartment

venting concept evaluated in NUREG/CR-0165 used a spray system with gravity feed to condense steam. These designs are different than proposed by UCS/NYPIRG, however, NUREG/CR-0165 and NUREG/CR-0138 (discussed below) were generic studies which provide an indication of the magnitude of consequence reduction which are possible. The designs proposed by UCS/NYPIRG should be able to be designed to ensure comparable risk reductions for Indian Point Units 2 and 3.

NUREG/CR-0138 reported risk reduction for a filtered venting system in terms of the reduction in probability of the WASH-1400 PWR release categories. These results indicate (NUREG/CR-0138, page 20): (a) a factor of 13 decrease in the probability of a PWR-1 release category (in-vessel steam explosion and missile penetration of containment); (b) a factor of 100 decrease in the probability of a PWR-2 release category (overpressure failure of containment); (c) a factor of 7.5 reduction in the probability of a PWR-3 release category (overpressure failure of the containment before core melt); and (d) a factor of 4.4 reduction in the probability of a PWR-4 release category (containment isolation failure).

NUREG/CR-0138 also evaluated consequence reduction resulting from implementation of a filtered venting system. With the WASH-1400 PWR assigned a value of 100, the report found that filtered venting would reduce early fatalities to 6.2% of their base value. In addition, latent cancer deaths would be reduced to 9.8% of their base value, and damage costs would be reduced to 26% of their base value (NUREG/CR-0138, page 23). It should be noted that this evaluation eliminated the interfacing LOCA containment failure mode, and assigned a failure

probability to the venting system of 1:100 (NUREG/CR-0138, page 21). Filtered venting system details were not addressed, but a factor of 100 reduction in elemental iodine and particulates was considered to be "reasonably achievable" (NUREG/CR-0138, page 5).

Finally, SAND80-0887, a status report on the Sandia filtered venting evaluation program (SAND80-0887, A.S. Benjamin and H.C. Walling, Development and Analysis of Vent-Filtered Containment Conceptual Designs, Sandia National Laboratories, undated, Attachment 5 to a letter dated 7 May 1980 from Milton S. Plesset, Chairman, ACRS, to NRC Commissioner Victor Gilinsky) contained risk curves to evaluate the impact of four filtered venting system options.[5] The evaluation was based on a TMLB' accident sequence in a 4-loop Westinghouse PWR at an unspecified location "chosen because of its proximity to a population center". The risk curves show elimination of early fatalities down to a conditional probability of 1:100 for all options evaluated. The risk curves show reductions of a factor of 10 (or greater) for latent cancer fatalities and the land area interdicted for more than 10 years for all four options evaluated. Other risk curves in NUREG/CR-1410 (pages 1.70 through 1.72) display similar results.

Q.25 What cost estimates exist for filtered venting and compartment venting systems?

A.25 We have performed no independent cost evaluations for the options proposed by UCS/NYPIRG. Early cost estimates for filtered venting and compartment venting systems presented in NUREG/CR-0165 (page 46) were up to \$10 million for a filtered venting system and \$20-40 million for a compartment venting system. The \$10 million figure for a filtered

venting system was criticized in NUREG/CR-1029 (pages 53-55).

More detailed and recent cost calculations were reported in NUREG/CR-1410 (1980, page 1.77) for a variety of filtered venting options. These options began with a suppression pool and escalated to progressively more complex systems. The estimated costs ranged from \$12.4 million for a simple system with a Seismic II rating to 32.3 million for the most complex system with a Seismic I rating. These costs did not include special instrumentation inside containment for control logic, nor any costs associated with down time or licensing delays. The systems assumed the use of the normal 3-foot diameter purge system valve.

Finally, the FILTRA system intended for installation at Barseback is projected to cost about \$25 million (Robert Skole, "Swedish Utility Getrs Okay for First Filter-Vented Containments", Nucleonics Week, Vol. 23, No. 49, December 9, 1982, page 1).

Q.26 In summary, why do you believe that implementation of a filtered vented containment system or a compartment venting system is necessary at Indian Point Units 2 and 3?

A.26 We have shown that core melt accidents dominate risk to the public. We have also shown that to achieve significant risk reductions, proposed solutions must address accident consequences. These conclusions were shown to be applicable to Indian Point Units 2 and 3. We have selected two mitigating system design options which we believe are feasible and would lead to significant reductions in risk at Indian Point Units 2 and 3. We believe that the costs of such systems is in the range of a few

tens of millions of dollars, certainly less than one hundred million dollars.

FOOTNOTES

1. Risk can be defined in absolute terms as the product of probability and consequences, i.e., consequences per reactor-year. Although we believe this is an inadequate definition of risk as consequences have been traditionally calculated, we use this definition here because it has been frequently used by the NRC Staff.

2. The RSSMAP results are reported in NUREG/CR-1659, Vols. 1-4, for Sequoyah Unit 1, Oconee Unit 3, Calvert Cliffs Unit 2, and Grand Gulf Unit 1, respectively. The IREP results are reported in NUREG/CR-2515 for Crystal River Unit 3, NUREG/CR-2787 for Arkansas Nuclear One Unit 1, and NUREG/CR-2802 for Browns Ferry Unit 1.

3. There has been criticism of the WASH-1400 consideration of external events such as earthquakes and hurricanes. In addition, the RSSMAP and IREP reports excluded external events as possible core melt accident initiators, focusing instead on events internal to the facility.

4. Draft, Letter Report on Review and Evaluation of the Indian Point Probabilistic Safety Study, Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, 25 August 1982.

5. Release Category 2RW, described in IPPSS at page 5.6-3 as dominated by an accident sequence involving core melt initiated by a loss of all AC power and loss of secondary heat sink, resulting in eventual containment failure by overpressure at 10 to 12 hours.

6. These measures are detailed in Appendices B and C of DD-80-5, "Director's Decision Under 10 CFR 2.206", dated 11 February 1980, U.S. Nuclear Regulatory Commission.

7. NUREG/CR-1410, SAND80-0617/1, W.B. Murfin, et. al., Report of the Zion/Indian Point Study: Volume I, Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, August 1980; NUREG-0850, Vol. 1, J.F. Meyer, et. al., Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects, U.S. Nuclear Regulatory Commission, November 1981; and IPPSS, Vol. 7, Section 3, "Degraded Core Phenomena", PASNY and Consolidated Edision Company of New York, March 1982.

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NUCLEAR REGULATORY COMMISSION '82 DEC 30 A11:44

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

In the matter of CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point, Unit No. 2) POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	Docket Nos. 50-247 SP 50-286 SP 29 December 1982

CERTIFICATE OF SERVICE

I hereby certify that single of: (a) UCS/NYPIRG TESTIMONY OF GORDON R. THOMPSON AND STEVEN C. SHOLLY ON COMMISSION QUESTION TWO, CONTENTIONS 2.1(a) AND 2.1(d); (b) UCS/NYPIRG SUPPLEMENTAL RESPONSE TO LICENSES' FIRST SET OF INTERROGATORIES UNDER COMMISSION QUESTION ONE; and (c) RESPONSE TO CONSOLIDATED EDISON'S MOTION TO ELIMINATE CONTENTIONS, were served upon the following by deposit in the U.S. mail, first class postage prepaid, this 29th day of December 1982, except where noted otherwise by asterisks.

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Steven C. Sholly

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* Documents served by messenger on 29 December 1982.

** Document (b) served by deposit in Express Mail on 24 December 1982; documents (a) and (c) served by deposit in Federal Express on 28 December 1982.

*** Document (b) served by hand on 27 December 1982 due to building being locked and preventing service on 24 December 1982; documents (a) and (c) served by hand on 28 December 1982.