UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

DIRECT TESTIMONY OF FRANK ROWSOME AND ROGER BLOND
CONCERNING COMMISSION QUESTION 5
A. COMPARATIVE RISK

Q.1 Mr. Rowsome, please state your name and position with the Nuclear Regulatory Commission, and describe your responsibilities in that position for the record.

A.1 My name is Frank H. Rowsome, I am Assistant Director for Technology, Division of Systems Technology of the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. In that position I manage the work of the Safety Program Evaluation Branch and of the Reliability and RISK Assessment Branch.

Q.2 Have you prepared a statement of your professional qualifications?

A.2 A statement of my professional qualifications as corrected was bound into the record of this proceeding following Tr. pg. 7169 February 9, 1983.

Q.3 Mr. Blond, please state your name and position with the Nuclear Regulatory Commission, and describe your responsibilities in that position for the record.

A.3 My name is Roger M. Blond, I am the Section Leader for the Accident Risk Section of the Reactor Risk Branch of the Division of Risk Analysis of the Office of Research. In that position, I am responsible for providing technical and managerial direction in developing methods and research in accident risk analysis and in performing applications in risk assessment.

Q.4 Have you prepared a statement of your professional qualifications?

A.4 A statement of my professional qualifications was bound into the record of this proceeding following Tr. pg. 7169 February 9, 1983.

Q.5 What is the purpose of this testimony.?

A.5 The purpose of Section A of this testimony is to provide the NRC Staff response to Commission Question 5, which reads

> "Based on the foregoing, how do the risks posed by Indian Point Units 2 and 3 compare with the range of risks posed by other nuclear power plants licensed to operate by the Commission? (The Board should limit its inquiry to generic examination of the range of risks and not go into any site-specific examination other than for Indian Point itself, except to the extent raised by the Task Force)"

This section of testimony will also address NRC Staff views on the significance of the high population density surrounding the Indian Point site to the comparative risk issue.

Section B of this testimony addresses the perspectives on acceptability of the risks posed by Indian Point.

Section C evaluates the regulatory actions considered by the Staff as a result of inferences drawn from the Staff's risk analysis and makes recommendations concerning such actions.

Q.6 Mr. Blond, please describe your role in the preparation of this testimony.

A.6 I am responsible for the quantitative comparisons made by the Staff between the Indian Point Probabilistic Safety Study and other published PRAs, and the site comparison in Section A of this testimony.

Q.7 Mr. Rowsome what was your role in the preparation of this testimony?

A.7 I am responsible for the qualitative comparison drawn between Indian Point and other reactors licensed by the Commission, as well as for the entire contents of sections B and C of this testimony.

Q.8 What conclusions were reached in the Indian Point Task Force Report concerning the comparison of the risk of Indian Point to other nuclear power plants?

A.8 In the spring of 1980, a snapshot was taken of the risk at Indian Point. This picture represented a very brief analysis which enabled a preliminary comparison to be made between Indian Point risks with risks of other reactor sites and designs. NUREG-0715 presented this comparison. The Task Force concluded that the Indian Point design was about ten times better than the average reactor design, but the site was ten times worse than the average site. Therefore, based upon this conclusion, Indian Point was not thought to pose undue public risk.

Q.9 What has happened since the Task Force report?

A.9 From the spring of 1980 to the present, a significant amount of information has become available which bears on the question of comparison of the risk at Indian Point with other plants. While the Staff was developing its estimates of the risk of Indian Point, Con Ed and PASNY published the IPPSS. The Task Force analysis presented in NUREG-0715 represents the briefest of PRAs, whereas the IPPSS and the current staff analysis represent the most comprehensive.

Second, there now exist significantly more PRAs than were available in 1980. Where there were six or seven PRAs available for comparison in 1980, there have now been at least 13 or 14 PRAs published. However, having this more comprehensive data base available has posed as many new questions as it has helped to answer. Questions concerning expanded scope, consistency of approach, adequacy of data, level of detail, and quality assurance are being raised as potential areas where comparison between studies could be faulted. Third, the state-of-the-art of probabilistic risk assessment is evolving continuously. Large amounts of resources in both private and government research are being expended to improve PRA methods and data. For example, external hazards and common cause events are currently being highlighted as being extremely important to a complete probabilistic risk assessment and a great deal of effort is being expended to improve the methods and data associated with these events.

Q.10 How will the comparison of Indian Point with other reactors be analyzed?

A.10 The comparison of Indian Point to other power plants will be divided into two parts: first, reactor designs will be compared; then reactor sites will be compared. For the design comparison, the staff has compared the results of the PRAs as published with no modification. It was felt that the published results represent the only consistent reference point for comparison purposes. Thus, the published IPPSS results unperturbed by current perspectives have been used in the design comparison. Currently, there have been PRAs published and available for at least the following 13 U.S. power plants.

		PRA/ Sponsor	Document	Yr of Pub.	NSSS	AE	Power MWe
1)	ANO-1	IREP/NRC	NUREG/CR-2787	81	B&W	Bechtel	836
2)	Big Rock	Consumers Power	-	81	GE	Bechtel	71
3)	Calvert Cliffs	RSSMAP/NRC	NUREG/CR-1659	82	CE	Bechtel	850
4)	Crystal River	IREP/NRC	NUREG/CR-2515	80	B&W	Gilbert	825
5)	Grand Gulf	RSSMAP/NRC	NUREG/CR-1659	81	GE	Bechtel	1250
6)	IP 2	IPPSS	-	82	W	UE&C	873
7)	IP 3	IPPSS	-	82	W	UE&C	965
8)	Limerick	Phil Elec	-	81	GE	Bechtel	1055
9)	Oconee	RSSMAP/NRC	NUREG/CR-1659	80	B&W	Bechtel	886
10)	Peach Bottom	RSS/NRC	WASH-1400	75	GE	Bechtel	1065
11)	Sequoyah	RSSMAP/NRC	NUREG/CR-1659	78	W	TVA	1148
12)	Surry	RSS/NRC	WASH-1400	75	W	S&W	775
13)	Zion	ZPSS	-	81	W	S&L	1100

These 13 units represent the full range of designs that have been built in the U.S. Caution must be exercised when using these results since there are very large uncertainties in these analyses. No attempt has yet been made to adjust the results to compensate for inconsistency of approach or methods. Therefore, the appropriateness of the comparison is in question. However, all of the studies have analyzed, in roughly the same manner, the so-called "internally" initiated events.

Q.11 What is meant by internally initiated events?

A.11 All reactor accidents can be classified as being either internally initiated or externally initiated. In conventional parlance, the phrase "internally" initiated is applied to spontaneous loss-of-coolant accidents, to spontaneous transient events, and to loss-of-coolant accidents induced by spontaneous transients. Commonly, but inaccurately, loss-of-offsite power is lumped with "internally" initiated events.

"Externally" initiated events include earthquakes, storms, floods, and fires, including inplant fires and floods.

Of the 13 PRAs, only 4 have included a risk assessment incorporating "externally" initiated events, and of these only the Zion Probabilistic

Safety Study approaches the completeness with which the IPPSS dealt with them.

Four figures have been prepared to compare the results of these PRAs. The results are taken directly from the published reports. The results have not been altered to reflect revisions, subsequent plant alterations or peer-review comments. The first three figures are relevant only for the internally initiated events for which there is the greatest confidence in performing such a comparison.

Q.12 What is compared in the first figure?

A.12 The first figure shows the expected internally initiated ("internal") core melt frequencies for the plants. The frequencies for the Indian Point units are shown first followed by the frequencies for the other units, presented from the lowest to the highest.

Q.13 How do "internal" core melt frequencies compare?

A.13 The "internal" core melt probabilities for the Indian Point units are about average with respect to the other plants.

Q.14 What is shown in the second figure?

A.14 The second figure shows the uncertainty bounds on the "internal" expected core melt frequencies as given in Figure 1. These uncertainty bounds represent about a 10 to 90% confidence interval* for

* The uncertainty bounds generally reflect statistical uncertainties only, not modeling approximation or completeness issues. Thus, the full uncertainty range is probably broader than indicated.

the expected core melt frequencies. Whenever estimates were given in the PRA for the magnitude of the uncertainty bounds, solid lines delineated by short lines at the top and bottom were used in the figure. When no estimates of the uncertainties were given in the analysis, staff estimates are shown with dashed lines delineated by question marks (?). As published, the smallest uncertainty band is about one order of magnitude, whereas the largest uncertainty band is over three orders of magnitude.

Q.15 How do the uncertainties of the internal events for core melt of Indian Point compare to the other plants?

A.15 As with the expected core melt frequencies, the Indian Point uncertainty bounds are within the range of uncertainty bounds of the other power plants.

Q.16 What is compared in the third figure?

A.16 Core melt frequency is an important measure of the plant vulnerability. However, it is not necessarily a measure of public health and safety risk. The third figure presents the estimated frequency of a severe release of radioactive material to the environment from internally initiated events. A severe release is defined as a release which has the potential to cause offsite doses which approach lethal levels (about 200 rem to the whole body).

Severe releases of radioactive materials are equivalent to the WASH-1400 release categories PWR 1, 2, or 3, or BWR 1, 2, or 3.

Therefore, they are roughly the same as Release Category A, B, or C events in the staff's Indian Point analyses. The frequency of severe release is a much more relevant measure of risk to the public health and safety when trying to compare the differences between plants. It is only from these releases that the public health and safety will be severely impacted. Therefore, the number of people in the vicinity of the site is only of critical importance should such a severe release occur.

Q.17 How do the "internal" severe release probabilities compare?

A.17 As given in the third figure, the Indian Point units are some of the best (lowest severe release frequencies) in comparison to the other power plants. This is a reflection of the capabilities of the engineered safety features at Indian Point with respect to internally initiated events. This was essentially the same conclusion that was arrived at in NUREG-0715. Unfortunately, the internally initiated events are not the end of the story concerning risk. In both Zion and Indian Point PRAs, "externally" caused events have been shown to be important contributors to risk.

Q.18 What is shown in the fourth figure?

A.18 The fourth figure compares the internally initiated events to the internally plus externally initiated events for those PRAs which included external analyses. As can be seen in the figure, the risk at Indian Point is totally dominated by externally initiated events.

It should be remembered that the external events analysis is the area where the methods and data are the most uncertain and, therefore, subject to the greatest error. Thus, extreme caution should be used in interpreting these results and making a comparison between plants based upon the "external" analysis of accident likelihood.

Q.19 Please elaborate on how the Indian Point Units 2 and 3 compare with other plants with respect to the likelihood of severe accidents originating in internally initiated events (loss-of-coolant accidents, transients, and loss of offsite power).

A.19 Virtually all reactor risk assessments performed to date have attempted to predict the likelihood that spontaneous loss-of-coolant accidents, transients, or loss-of-offsite power would result in core melt. Some studies have also attempted to calculate the quantities of radioactive material released in these accidents. The accident sequences that may potentially lead to high risk for pressurized water reactors like the Indian Point units have been found to be (1) uncontained interfacing systems loss-of-coolant accidents, (2) loss of all offsite and onsite AC power and failure of the auxiliary feedwater system, and (3) special plant-specific vulnerabilities. (Other potentially high risk scenarios originating in core melt phenomenology or containment response characteristics will be treated separately below.) As will be shown below, the Indian Point Units 2 and 3 are safer than average with respect to these three classes of accident susceptibility.

Q.20 Why are uncontained interfacing system loss-of-coolant accidents particularly important to reactor safety?

A.20 In an uncontained interfacing system loss-of-coolant accident, a set of valves at the pressure boundary separating the reactor coolant from a low pressure system outside containment is postulated to fail open. The low pressure system is assumed to rupture outside containment when exposed to the full pressure of the reactor coolant system. This results in a loss-of-coolant outside containment. Such an accident would be particularly severe because two valve failures give rise to three attributes: (1) it is a loss-of-coolant accident, i.e., a critical challenge to the emergency core cooling system, (2) the emergency core cooling system will fail in recirculation if not in the injection mode because the lost coolant does not accumulate in containment, from which it could be recirculated, so the core will almost certainly melt down, (3) containment is bypassed by the path of the rupture, so the fission products released from the melting fuel cannot be bottled up in the containment.

Such uncontained loss-of-coolant accidents are believed to be very unlikely at any plant, but their severity is such that they are believed to be among the dominant contributors to risk.

Q.21 Why are the Indian Point units less vulnerable to uncontained loss-of-coolant accidents than most reactors are?

A.21 Every pressurized water reactor plant is susceptible to uncontained loss-of-coolant accidents in the way the Indian Point units are, and most of the others are susceptible in other ways not shared by Indian

Point. Most pressurized water reactors have three to six large pipes that lead from the reactor to low pressure systems outside containment. The Indian Point Units have only one.

Second, the Indian Point units have pressure boundary valves on the one line potentially subject to uncontained loss-of-coolant accidents whose design is as reliable as any in nuclear power plants.

Q.22 Why are accidents entailing loss of all offsite and onsite AC power and loss of auxiliary feedwater particularly important to risk?

A.22 Virtually all of the ways of cooling the reactor fuel or of removing heat from the containment require electrical power to run motor-driven pumps. The one exception is the steam-turbine-driven auxiliary feedwater pump. Were all of these systems to fail and remain failed, then the core would melt, and the heat released by the fuel to the containment would accumulate until the containment ultimately fails.

Q.23 Why are the Indian Point units less likely to succum to a loss of all AC power and a failure of auxiliary feedwater than most other pressurized water reactor plants?

A.23 The principal advantage the Indian Point units have over other nuclear plants is the presence of three gas-turbine-driven electrical generators as well as three emergency diesel generators per unit to supply essential AC power requirements. No other reactor plant has so many backup sources of AC power. Many have only two diesel generators.

History has shown the Indian Point site to suffer complete losses of offsite power at roughly the same frequency as the national average. The typical outage duration may be somewhat longer than the average, but this is more than compensated for by the diverse and redundant alternative AC generators.

Q.24 What is meant by special, plant-specific vulnerabilities and why are they important to risk?

A.24 Among the risk assessments of other reactors, examples have been found of designs that met the licensing requirements, applicable codes and standards, etc., but which were nonetheless unnecessarily vulnerable to severe accidents of one kind or another.

An unnecessary vulnerability to severe accidents was found in the risk assessment of Sequoyah performed in the Reactor Safety Study Methodology Applications Program. In this case a simple human error in reverting from the refueling mode to the plant configuration for power generation might go undetected and compromise the ability of the plant to mitigate loss-of-coolant accidents. This vulnerability has been fixed.

This example indicates that risk assessments are capable of revealing some kinds of design deficiencies that have been missed by conventional licensing safety analysis and are important to risk.

Q.25 Why are the Indian Point units thought to be safer than average with respect to such design deficiencies?

A.25 The Indian Point units have been subject to one of the most thorough risk assessments ever done on a nuclear plant and no such plant-specific design deficiencies of importance to risk have been found for "internally" initiated accidents.

Q.26 What about the overall frequency of core melt?

A.26 Indian Point Unit 2 has a predicted median frequency of core melt of roughly 4×10^{-4} per year, or an average interval of 2500 years of service before a core melt would be expected, for the "after fix" staff analysis described in the testimony on Commission Question One. Indian Point Unit 3 also has a predicted frequency of core melt around 4×10^{-4} per year. Results from risk assessments of other plants range from 10^{-3} per year to about 10^{-5} per year, with many clustering near the values for Indian Point. This is not, however, a fair comparison for two reasons: (1) the Indian Point risk assessments are more comprehensive than the others. Had the Indian Point studies been limited to the same scope as the others, lower core melt frequencies would have been predicted. (2) Not all core melts contribute significantly to offsite radiological risk; many core melt sequences are well-contained. Thus only core melt accidents of severe consequences should be the basis for comparison. This is what we have done in Figure 3 and in the testimony above.

Q.27 Please list the ways the containment building might fail prior to core melt in an accident, indicate the importance to risk of these scenarios, and indicate how the Indian Point Units 2 and 3 compare with other plants with respect to each scenario.

A.27 The containment building might fail in an accident sequence prior to core melt for one of the following reasons:

1. Overpressure failure

   If a loss-of-coolant accident were to take place with initially successful core coolant replenishment but with failed containment atmosphere cooling, the steam pressure could gradually build up until the containment fails. Core cooling might well be disabled after containment rupture. Such scenarios have been found to be very important to the risk in some other plants. A number of design features make this scenario extremely unlikely at either Indian Point unit. Indian Point is well below average in its vulnerability to this class of accident sequences. (See also NUREG/CR-2934.)

2. Uncontained, interfacing system loss-of-coolant accidents

   These were discussed above.

3. Earthquake-induced collapse of containment

   The Indian Point Probabilistic Safety Study found that extremely severe, extremely rare earthquakes might induce containment collapse. This scenario is one of the principal contributors to the predicted risk of early fatalities at Indian Point, according to the staff analysis, but not a dominant contributor to other offsite radiological risks. We do not know how likely such scenarios are at other plants.

Note also that the licensees have reassessed this vulnerability in their ammendment to the IPPSS. Licensee testimony on Commission Question 1 suggests that this failure mode is very much less likely than the original IPPSS suggested. The staff has not yet evaluated the validity of this reanalysis.

4. Internal missiles might breach containment prior to core melt.

   If the lid of the reactor vessel were to blow off, it might breach containment. All mechanisms evaluated to date for such failures indicate negligible probability and thus very small risk. There is no reason to believe that either Indian Point unit is above average in its susceptibility to any of these internal missile scenarios.

5. External missiles or explosions might breach containment, and possibly also cause core melt.

   Fragments from a disintegrating steam turbine, high speed crashes of very heavy aircraft, or very large external explosions might breach containment. These scenarios are believed to be too rare to contribute significantly to the risk at any plant. The Indian Point units are about average in their susceptibility.

6. Open containment penetrations

   A containment penetration might be open during an accident. This would have a leveling effect on core melt accident releases. Core melt accidents that would otherwise have been well-contained generate much larger releases if there is an open containment penetration. On the other hand, accident sequences which would otherwise have burst containment are made much more gradual: the plume would contain much lower rates of fission product release than would a puff-release. Open penetrations have been found to play a very small role in the safety profile of the Indian Point Units.

7. Steam generator tube rupture with a stuck-open atmospheric steam relief valve.

   A path by which fission products might bypass containment and escape to the atmosphere can occur if a steam generator tube ruptures and a valve on the associated main steam line is open to the atmosphere. This scenario resembles the uncontained interfacing system loss-of-coolant accident insofar as it entails opening a path from the reactor coolant system to the outside atmosphere. However, this scenario is very much less serious for a variety of reasons. First, core melt is not assured; it is extremely unlikely. The operators can terminate the loss of coolant by depressurizing and cooling the reactor. They have a long time in which to do this. The operators also have ample opportunity to try to close the failed-open steam relief valve. Bases for the conclusion that steam generator tube rupture accidents are not characterized by particularly severe consequences can be found in Answer 13 B of the staff testimony on Board Question 2.2.1. Therefore, this scenario is thought to make a very small contribution to the risk at any pressurized water reactor. The Indian Point units are average in their susceptibility to such scenarios.

Q.28 Please list the ways the containment might fail during core meltdown, indicate the importance to risk of these scenarios, and indicate how the Indian Point units compare with other plants for each scenario.

A.28 A reactor containment might fail at the time of core meltdown from one of two mechanisms, a pressure spike or internal missile.

1. Pressure spike

The pressure within the containment may rise sharply at the time of reactor vessel melt-through from one or a combination of effects: (a) compressed gasses released from the reactor coolant system, (b) steam generated if the molten core falls into a pool of water in the reactor cavity, or (c) burning of hydrogen in the containment atmosphere ignited by the molten core material.

Dr. Meyer has described in Section III.B his calculations which indicate that these mechanisms will not fail the Indian Point containment buildings. PRAs of other plants have suggested that this may be important to risk at other plants, particularly those with smaller or weaker containment buildings. Most of these other studies are more pessimistic and less thorough than Dr. Meyer's analysis of Indian Point. It may be that the analyses of other plants are unduly pessimistic. In any case, the Indian Point units are among the better plants at containing the pressure spikes associated with reactor vessel melt-through.

2. Reactor vessel missiles

A mechanism has been postulated by which the core meltdown process might generate reactor vessel missiles that might breach containment. It is a steam explosion as the molten core slumps into the water remaining in the lower hemisphere of the reactor vessel. Molten core material poured into water sometimes gives rise to explosive boiling of the water. Recent theoretical and experimental analyses suggest that steam explosions can take place but never approach the energy needed to burst the reactor vessel.

There is no reason to believe that the Indian Point units are any more susceptible to this containment failure scenario than any other plant.

Q.29 Now let us turn to the effectiveness with which an initially intact containment can bottle up core melt accidents. Consider first those scenarios in which one or both of the containment heat removal systems is operable or is restored to function during the accident. How well is the containment predicted to perform at Indian Point, how important are these scenarios to risk, and how do the Indian Point Units 2 and 3 compare with other plants for such accidents?

A.29 As Dr. Meyer has testified in Section III.B of the Staff's testimony on Commission Question 1 if either or both containment heat removal system is functional, the containment should be successful in bottling up the fission products released to the containment atmosphere. Only a very slight leakage is predicted and the offsite radiological risk is negligible.

The containment may be challenged by the possible burning of hydrogen in the containment atmosphere, Dr. Meyer's analysis suggests that hydrogen burns are unlikely to fail containment. If it did fail, the release would be of intermediate severity. Since such failures are thought to be unlikely, the risk contribution is low.

The containment basemat might also be melted through by the core debris. Dr. Meyer's analysis suggests that this is unlikely and would take three days or more. In any case, were this to happen, the ground would be an effective filter for the particulates released. The effects of the airborne plume would be very minor. See also testimony in IV.A.1. Radioactive materials introduced to the groundwater following basemat melt-through can probably be intercepted before they reach the Hudson River, and would not be particularly severe even if they reached the Hudson. See also Richard Codell's testimony in Section III.D.

In short, core melt accidents in an initially intact containment, having one or more operable containment heat removal systems are expected to be well contained and to pose negligible offsite radiological risk.

Plants with smaller and/or weaker containments than Indian Point are thought to be less reliable at containing such core melt accidents. The Indian Point units are believed to be among the better at mitigating the consequences of these classes of accident sequences.

It should be noted, however, that we are not completely confident that the containment heat removal systems can continue to function for a long time after a core melt accident. Core debris particles might foul the containment spray recirculation system. Fine particulates in the containment atmosphere might foul the filters or cooling coils of the containment air coolers. The experimental evidence is ambiguous. The Sandia review of IPPSS documents a sensitivity study on this issue in NUREG/CR-2934.

Although we do not have a definitive answer on the operability of containment heat removal systems after core melt, we can say these things about it:

1. Even if the containment heat removal systems are always defeated by core melt accidents, the percentage increase in predicted risk for Indian Point would not be large. See Section III.B of the Staff testimony on Commission Question One for a sensitivity study on this question done by Dr. Meyer.

2. Indian Point Unit 3 may have an advantage over Indian Point Unit 2 in this respect. The filters associated with the containment air coolers (fan coolers) will be automatically bypassed in the event that they are plugged with particulates. This is not the case for Unit 2. Thus the Unit 2 coolers may be more vulnerable to fouling than those of Unit 3.

3. The location and geometry of the containment air coolers and the emergency sump are better isolated from regions likely to be fouled by core debris than is the case at many pressurized water reactors. Thus they are likely to be less vulnerable to containment heat removal failure due to core melt debris than the average plant.

Q.30 How effective are the Indian Point containments at mitigating the effects of accidents in which both core cooling and containment heat removal fail, but the containment is initially intact?

A.30 In accident sequences in which a common root cause defeats both core cooling and containment heat removal functions, the core will melt down and the pressure in the containment building will gradually rise. Ultimately the containment may fail in one of three ways, or it may succeed in bottling up the event.

(1) It may rupture on overpressure

(2) It may begin to leak at a rate that limits the pressure increase

(3) The core may melt through the basemat, thus relieving the pressure, before one of the other two failure modes takes place.

(4) The containment may remain intact with little or no leakage.

The staff testimony did not include an estimate of the likelihood that a small, gradual leak might develop and thus head off an overpressure rupture. Dr. Meyer has been able to calculate the fraction of these scenarios in which overpressure failure precedes basemat melt-through. This model may be summarized as follows:

(1)	Overpressure rupture	40%
(2)	Gradual leakage	unknown
(3)	Basemat melt-through	53%
(4)	No failure	6%

Dr. Meyer's most realistic estimate of the time to basemat failure from the time of core melt is roughly three days. This mode of failure and time of failure is for the dry cavity case. However, in some variants of the accidnet sequences, the cavity may have sufficient water in it to dail the containment by overpressurization is as little as helf a day. Thus the time of failure varies from as little as 13 hours, for the overpressure rupture mode, to 3 days or longer for basemat melt-through.

The time to release is important to the severity of the radiological release in several ways. Not only does more time allow for more reliable evacuation but also the quantities of radioactive materials that ultimately escape diminish, in part because of radioactive decay--some of the most hazardous radioisotopes have very short half lives--and in part because radioactive particulates or gasses soluble in water have more opportunity to fall out or plate out inside containment. Thus the quantity of hazardous materials in the containment atmosphere potentially available for release decline with time.

Q.31 How important to the risk profile of Indian Point plants are the accidents in which failure of both core cooling and containment cooling lead to overpressure failure of containment?

A.31 A large part of the offsite radiological risk projected for the Indian Point units originate in these accidents. The risk dominant earthquake, fire, and storm scenarios result in failures of control power supplies or bulk ac power supplies needed for both core cooling and containment heat removal. Forty percent of these accidents are postulated to result in overpressure failure of containment. These, in turn, contribute well over 90% of the expected risk of early injuries, and virtually all (99% or more) of the latent casualties, and offsite property damage. These scenarios are also important to early fatalities.

Q.32 How important to the risk predictions for Indian Point are more than 50% of the loss-of-all-cooling scenarios that result in basemat melt-through?

A.32 As noted above, basemat melt-through produces negligible offsite radiological effects. Virtually none of the risk--except in the arena of onsite property damage--is contributed by basemat melt through.

Q.33 Suppose we were to discover that in most of these scenarios the containment starts to leak as the pressure approaches rupture pressure, so that rupture never takes place. What effect would this have on the risk predictions?

A.33 Leakage releases less of the containment atmosphere and does so more gradually than overpressure rupture is predicted to do. The licensees' testimony on Commission Question One adopts this model.

Q.34 How do the Indian Point Units 2 and 3 compare with other plants in the risk posed by the loss-of-all-cooling accidents leading to overpressure failure of containment?

A.34 The Indian Point containment design is superior to most other large dry containments at delaying overpressure failure and affording an opportunity for the benign basemat melt-through to precede and prevent the far more hazardous overpressure failure. This advantage in the Indian Point design originates in large containment volume in proportion to the reactor power level, high failure pressure, and the use of basaltic concrete in the basemat. In no other reactor plant studied to date is basemat melt-through predicted to occur before (and thus prevent) overpressure failure in a core melt accident with all containment heat removal systems disabled. It should be noted, however, that these prior studies of containment response were less thorough than the Staff analysis of

Indian Point. In deference to the lower accuract of the other studies, more pessimistic assumptions were used. It is plausible that Indian Point is not alone in being able to effectively mitigate some loss-of-all-cooling accidents.

Still, the large containment volume, high failure pressure, and the basaltic concrete basemat suggest that the Indian Point units are better than the average large dry containment plant at containing loss-of-all-cooling accidents.

Some plants utilize passive devices to capture steam and heat in the containment atmosphere; i.e., ice or pools of water. These require no actuation signal or power supply to perform their function. Designers of reactor plants with passive steam condensation devices take advantage of them to employ smaller and/or weaker containment buildings. Their heat absorption capability is finite and will be overwhelmed in time if no active heat removal system operates to dissipate the reactor after-heat to the environment. Such containments are also predicted to fail due to overpressure in loss-of-all-cooling accidents. They are less able to bottle up the non-condensible gasses generated in core meltdown or core-concrete interactions.

In light of the limited ability of the pressure-suppression containment designs to contain the non-condensible gasses associated with core meltdown and core-concrete interactions, we believe them to be no better and possibly not as good as the Indian Point containment design at mitigating loss-of-all-cooling accidents. A quantitative comparison cannot be made because the risk assessments of these other containment designs completed to date are not comparable to Dr. Meyer's analysis of Indian Point; they are of the WASH-1400 style--less thorough and more conservative.

Note that while the Indian Point containment design is thought to be one of the best at mitigating loss-of-all-cooling accidents, such accidents may be more probable at Indian Point than at some--perhaps many--other plants. The comments about the comparative likelihood of severe release accidents at the beginning of the section of testimony apply to the loss-of-all-cooling accidents.

Q.35 Please summarize the assessment of the comparison of the risk of Indian Point with other reactors in terms of the design.

A.35 Both Indian Point units are less likely than the average plant to have large releases of radiation due to the accident sequences studied in most PRAs. This group of accident sequences is triggered by loss-of-coolant accidents, transients, and loss of offsite power. We do not know whether they are above average, average, or below average in the likelihood that earthquakes, fires, or storms may cause severe accidents, because we have little basis for comparison. The containment employed at Indian Point is among the best in the effectiveness with which it can mitigate the offsite radiological releases given a core melt, although this may be an articact of the analytical methods. On qualitative grounds we believe the containment to be among the more effective designs.

Q.36 In what ways are the characteristics of the Indian Point site atypical, and how do these characteristics affect risk?

A.36 As was discussed in NUREG-0715, by far the most important difference between the Indian Point site and the average nuclear power plant site is the population density in the surrounding area. The population within a circle around the site of 10, 30, and 50-mile diameter are each about 10 times greater than the population around the median site. Because of the high population densities, the site is also atypical in the property values of the land around the site. The weather at the site is entirely typical, so the site is quite average in the risk posed to individuals, insofar as site variables are concerned. The speed and reliability of evacuation could in principle--affect individual risk. However, we expect the risk to individuals to be quite insensitive to evacuation speed. See also the staff testimony on Commision Question One, Part IV B.

In short, if the same plant were located at a typical site, the risks to individuals would be the same as they are, but the societal risks would be about ten times smaller, simply because there are roughly ten times as many people at risk around the Indian Point site as there are at an average site.

Q.37 Explain the relationship between the Indian Point site and the other reactor sites as described in NUREG/CR-2239, "Technical Guidance for Siting Criteria Development."

A.37 NUREG/CR-2239 presents technical information relevant to support the formulation and comparison of possible generic siting criteria for nuclear power plants. Four areas were investigated in the study. These are: consequences of hypothetical severe nuclear power plant accidents; demographic characteristics about current reactor sites; site availability; and socioeconomic impacts of siting.

The consequences of hypothetical reactor accidents were calculated using a representative set of severe accident source terms which covers the full spectrum of releases that have been postulated in previous PRA studies.

As was discussed in detail in the report, three generic siting source terms (SSTs) were used to scope the siting question and provide insight into the relationship of the source term to the site characteristics. The absolute and relative probabilities associated with the accident source terms used in the study are generically based and are not a reflection of a specific design or a given facility.

The SSTs were used to calculate accident consequences at 91 U.S. reactor sites using site specific meteorology and population data and assuming an 1120-MWe reactor. The CRAC2 computer code was used to calculate the consequences. Siting factors such as meteorology and emergency planning were also investigated and sensitivity studies were conducted for most of the parameters important to siting considerations.

Appendix C of the study presents site-specific consequence estimates for each of the three siting source terms. It is important to note the assumptions used in the calculations: (1) standard 1120-MWe PWR; (2) summary evacuation; (3) actual site population and wind rose; and (4) best estimate meteorology. Conditional complementary cumulative distribution functions are given for each site using the SST1 (largest source term) accident release for early fatalities, early injuries; and latent cancer fatalities. Mean numbers have also been tabulated for each of the SST source terms and consequence groups given above. The range of mean early fatalities for SST1 is from 0.07 for the Vogtle site to 970 for the Limerick site. The median site is predicted to have about 30 mean early fatalities. Indian Point, with 830 mean early fatalities is less than a factor of 30 greater than the median site for SST1. Mean early injuries range from 3.7 at Pebble Springs to 3600 at Indian Point. Indian Point was calculated to be about a factor of 15 greater than the median site. Mean latent cancer fatalities for SST1 range from 310 at WPPSS to 8100 at Indian Point. Indian Point was found to be about a factor of 6 higher than the median site.

When SST2 is considered as the representative accident source term, the absolute magnitudes associated with the consequences are appreciably reduced. The maximum number of mean early fatalities, injuries, and latent cancer fatalities for any site is 2, 18, and 590, respectively. Thus, the variation between sites is significantly reduced when compared to SST1.

For the SST3 source term, there is almost no substantive difference between any site. Therefore, the source term questions will bear directly on the relationship between the Indian Point site and other reactor sites in answering the comparative risk question.

NUREG/CR-2239 offers the most comprehensive evaluation into the technical considerations of siting that has been performed to date. The report provides an information base to judge the relationship between all of the siting parameters. As evaluated in the report, the Indian Point site does have one of the largest numbers of people of any site in the country. However, the Limerick site outside of Philadelphia and the Zion site outside of Chicago are comparable to the Indian Point site in absolute numbers of people. The consequences calculated for Indian Point for the SST source terms are among the largest calculated for any site. However, there are other sites in the country with larger calculated consequences.

Figure C-8 on page C-14 of NUREG/CR-2239 (attached) is very informative in comparing the Indian Point site. Even for the SST1 source term, Indian Point and Limerick are almost identical in their predicted risk curves for all consequences. The LaSalle and LaCrosse sites are fairly typical of the average site in the country and the Kawaunee site is on the lower end of the consequence spectrum.

A final perspective on the comparative risk of Indian Point is given in Figure 2.4.2-1 on page 2-33 of NUREG/CR-2239 (attached).

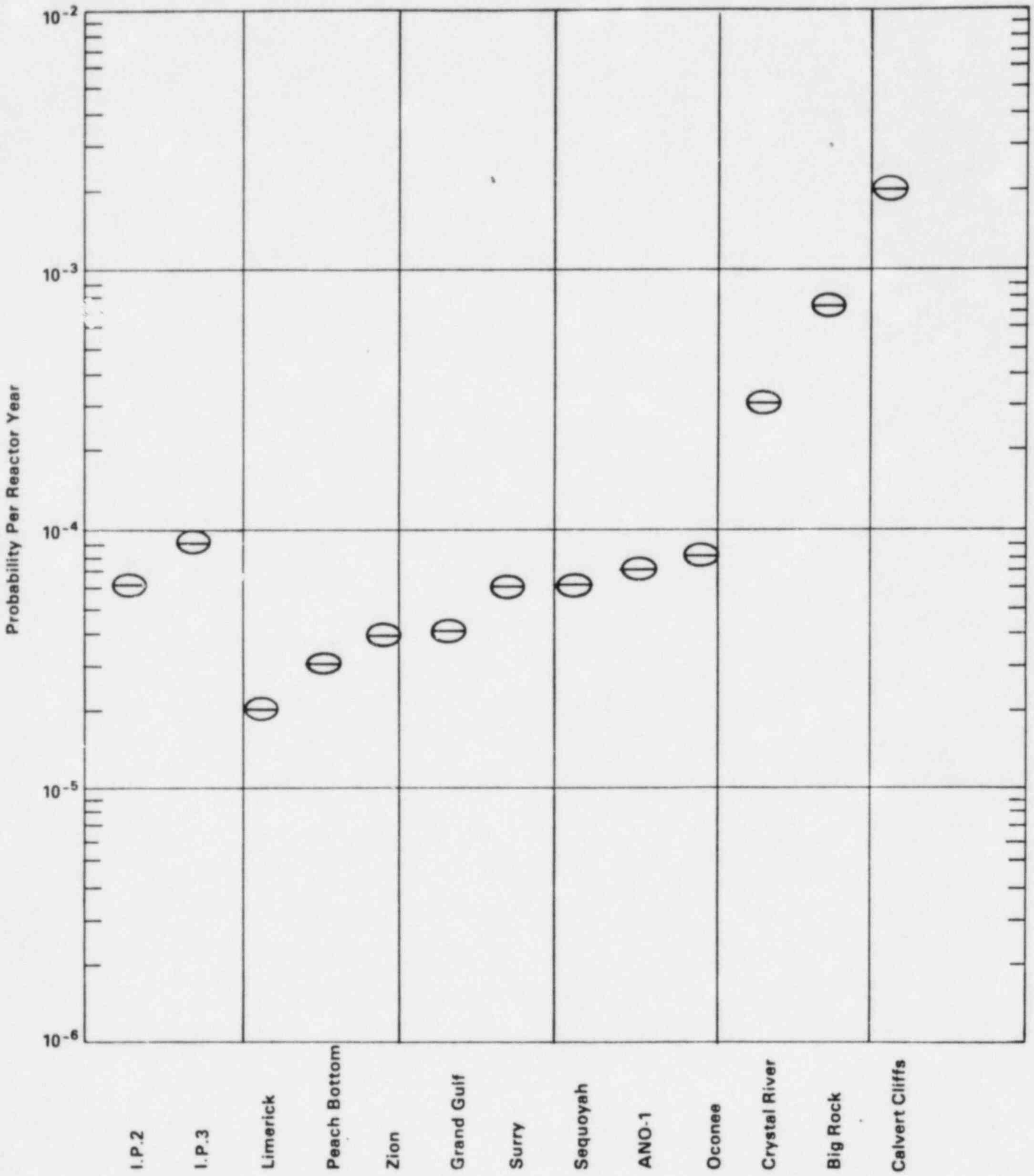
The figures on this page contain all of the 91 sites CCDF curves plotted together for early fatalities, early injuries, and latent cancer fatalities. For early fatalities there is about two and one-half orders of magnitude spread between the sites for the probability of killing at least one person. In addition, there is about a three and one-half order of magnitude spread in early fatalities at the one in a thousand conditional probability level. However, there are no clear discontinuities in the family of curves. That is, there are no apparent break points which would indicate that one site or even a group of sites are out of the reasonable bounds of the family of sites. Therefore, none of the sites can be said to represent a unique extreme in the continuum of current sites. Injuries and cancer fatalities further reinforce this notion, since the spread between sites is much smaller than for early fatalities. Thus even though the Indian Point site is one of the worst sites in terms of its demographic characteristics, it is not unique or even unusual as far as the set of sites is concerned.

The picture presented in the siting report is a perspective on siting factors only. As has been repeatedly stated in the report, the estimates presented are consequence estimates which can be used for comparative purposes only. They represent only the consequences of three hypothetical source terms which are not necessarily relevant to a specific reactor design.

Q.38 Please summarize how the Indian Point Units compare in severe accident risk to other nuclear power plants, all things considered.

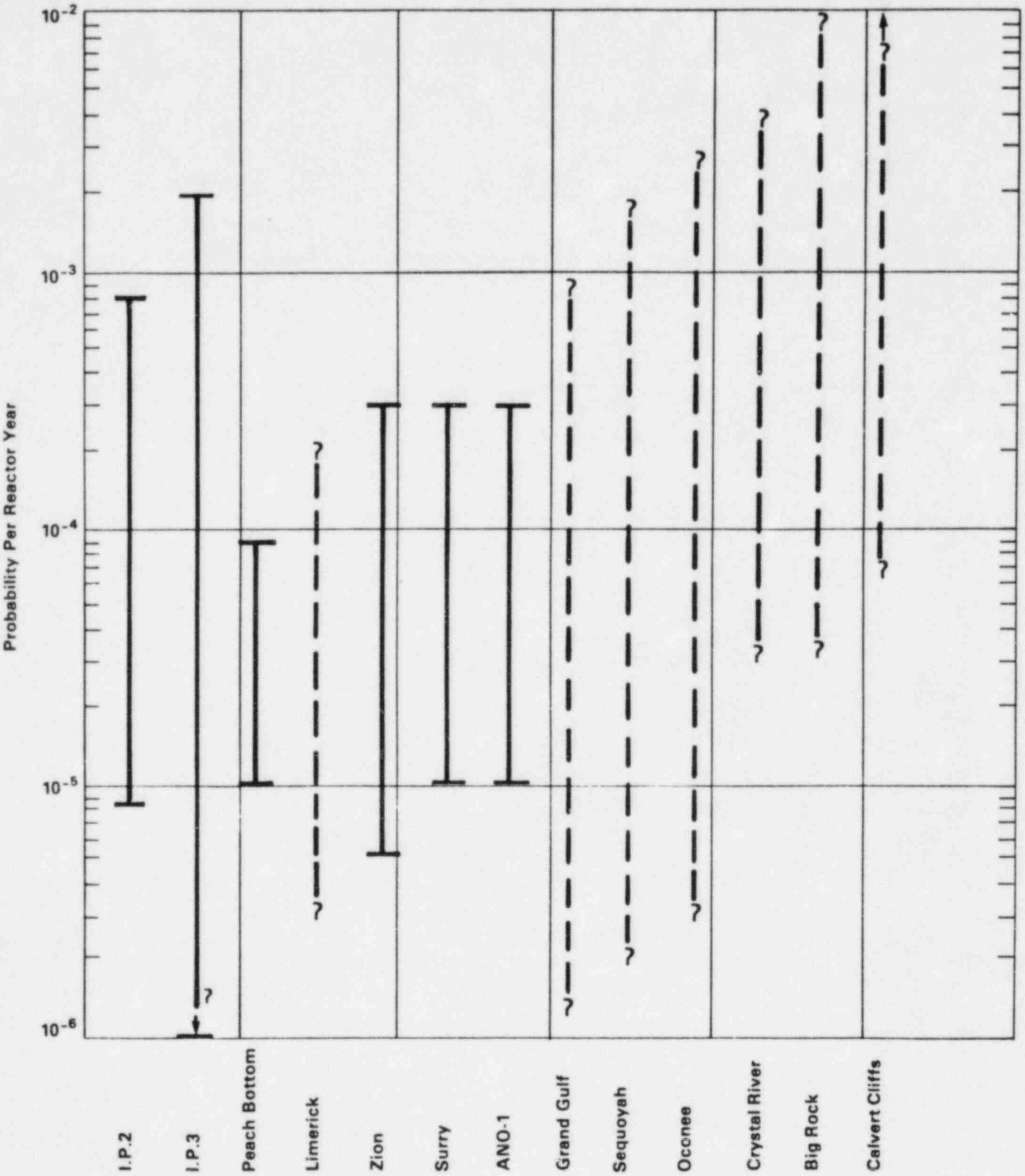
A.38 The Indian Point units are almost certainly not among those most likely to give rise to severe releases of radiation. They may be average, somewhat above average, or somewhat below average in the likelihood of severe releases. We are unable to tell because comparable assessments of the vulnerability of other plants to earthquakes, fires, and storms have not been done. The containments at Indian Point are thought to be better than average. The containment design affords a long warning time prior to overpressure failure, which is responsible for over 99% of the predicted occurrences of severe releases. The containment can successfully bottle up or produce benign releases in almost all the other core melt scenarios. The site is typical in individual risk characteristics and about ten times higher than average in population, and hence in site effects on societal risks. The net effect of these characteristics is ambiguous. Individual risks are probably average to well below average. Societal risks are probably average to above average. There is no reason to believe that either individual or societal risks are well outside the range or risks posed by other nuclear power plants licensed to operate by the Nuclear Regulatory Commission.

Figure 1 Expected Core Me.t Frequency for Internal Events ★



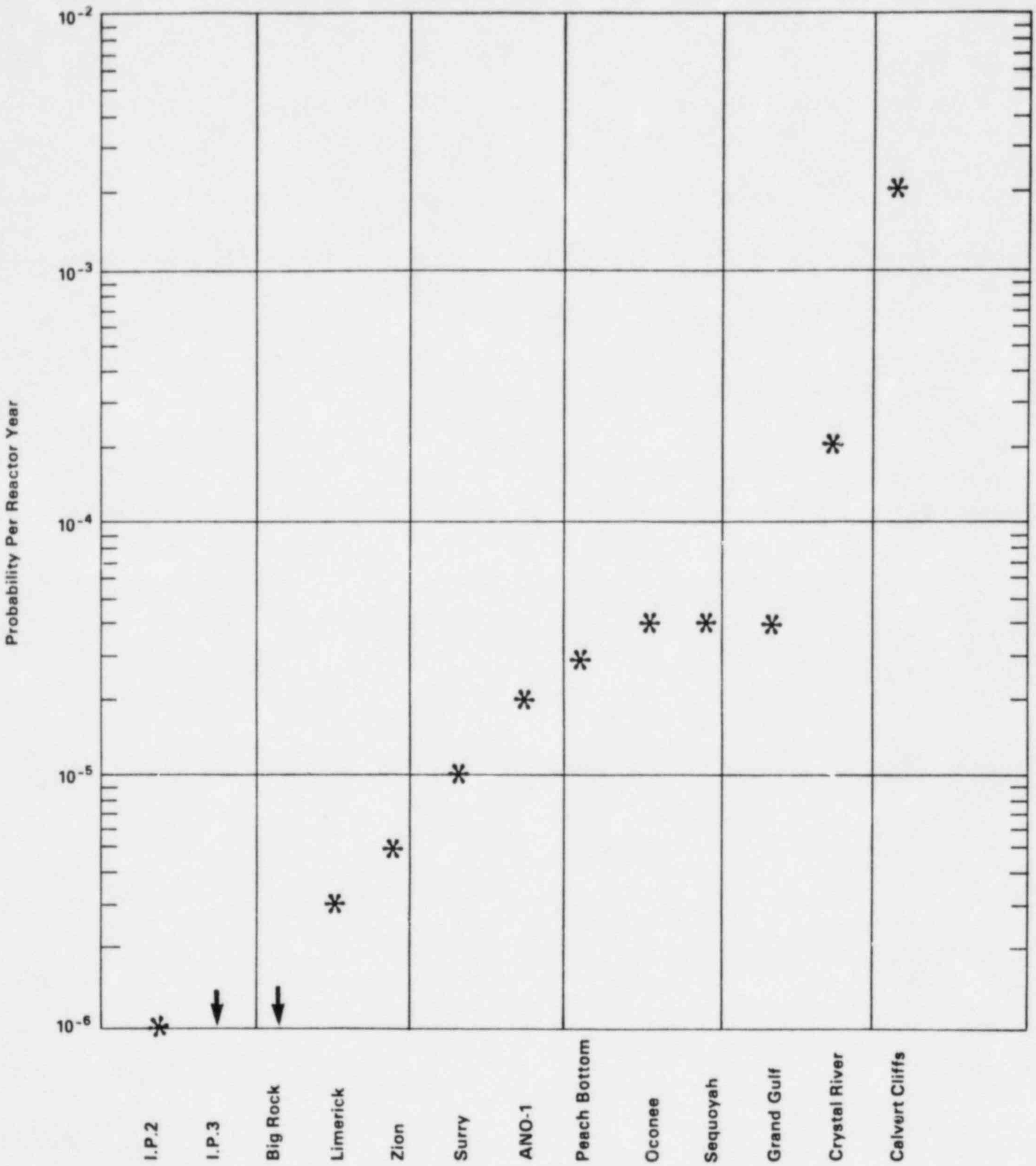
★ NOTE: Results presented o. this figure are taken directly from published PRAs without modification. The PRAs were not necessarily performed using consistent methodologies or assumptions. Many of the PRAs evaluate designs that have subsequently been altered.

Figure 2 Core Melt Uncertainty Bounds (90% — 10%) *
for Internal Events



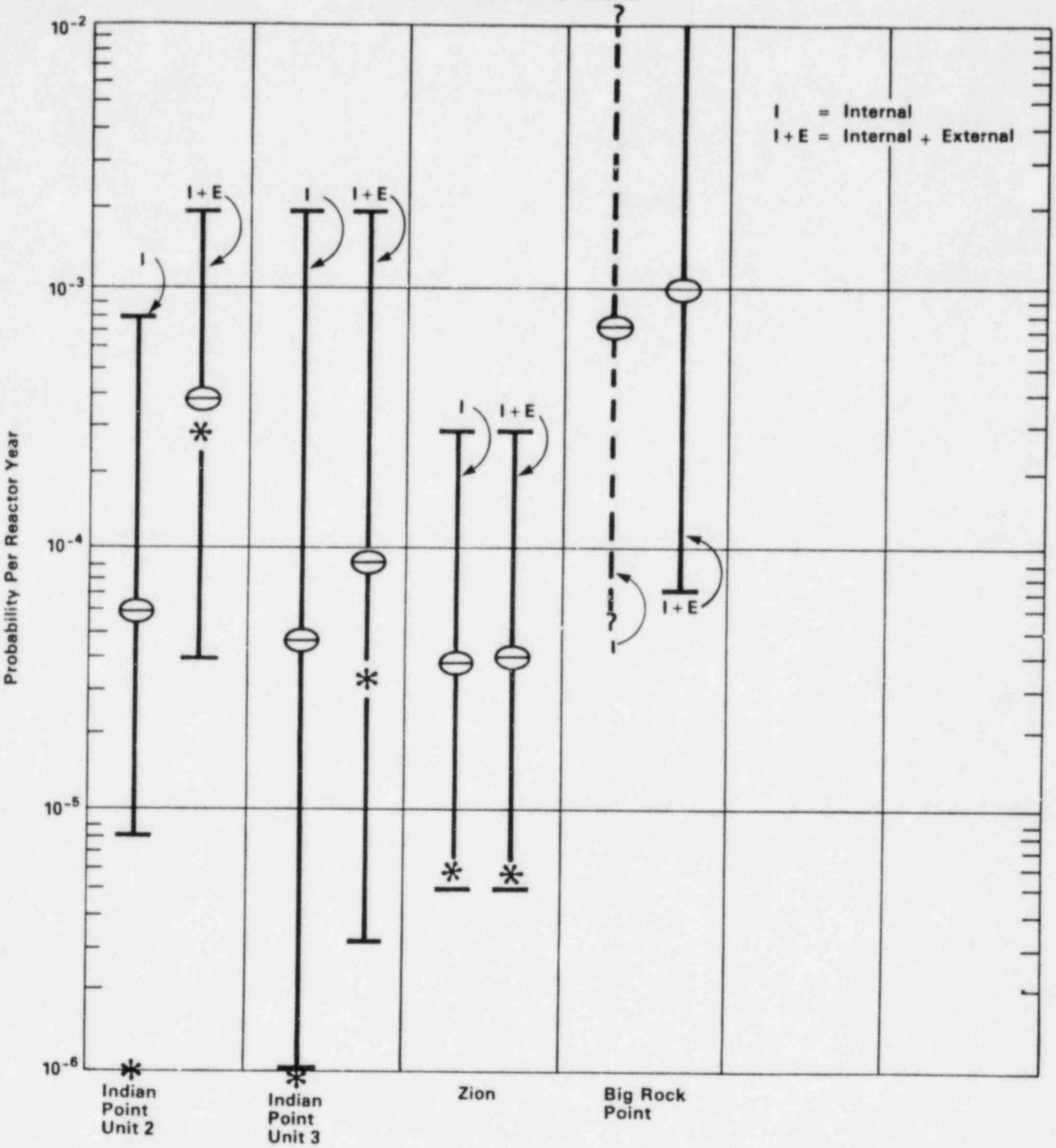
* NOTE: Results presented on this figure are taken directly from published PRAs without modification. The PRAs were not necessarily performed using consistent methodologies or assumptions. Many of the PRAs evaluate designs that have subsequently been altered.

Figure 3 Severe Release Probability for Internal Events *



* NOTE: Results presented on this figure are taken directly from published PRAs without modification. The PRAs were not necessarily performed using consistent methodologies or assumptions. Many of the PRAs evaluate designs that have subsequently been altered.

Figure 4 Uncertainty Bounds for Internal Events and * Internal Plus External Events



⊖ – Best Estimate Core Melt Probability

* – Best Estimate Sever Release

I – 10% – 90% Uncertainty Band About Best Estimate

* NOTE: Results presented on this figure are taken directly from published PRAs without modification. The PRAs were not necessarily performed using consistent methodologies or assumptions. Many of the PRAs evaluate designs that have subsequently been altered.

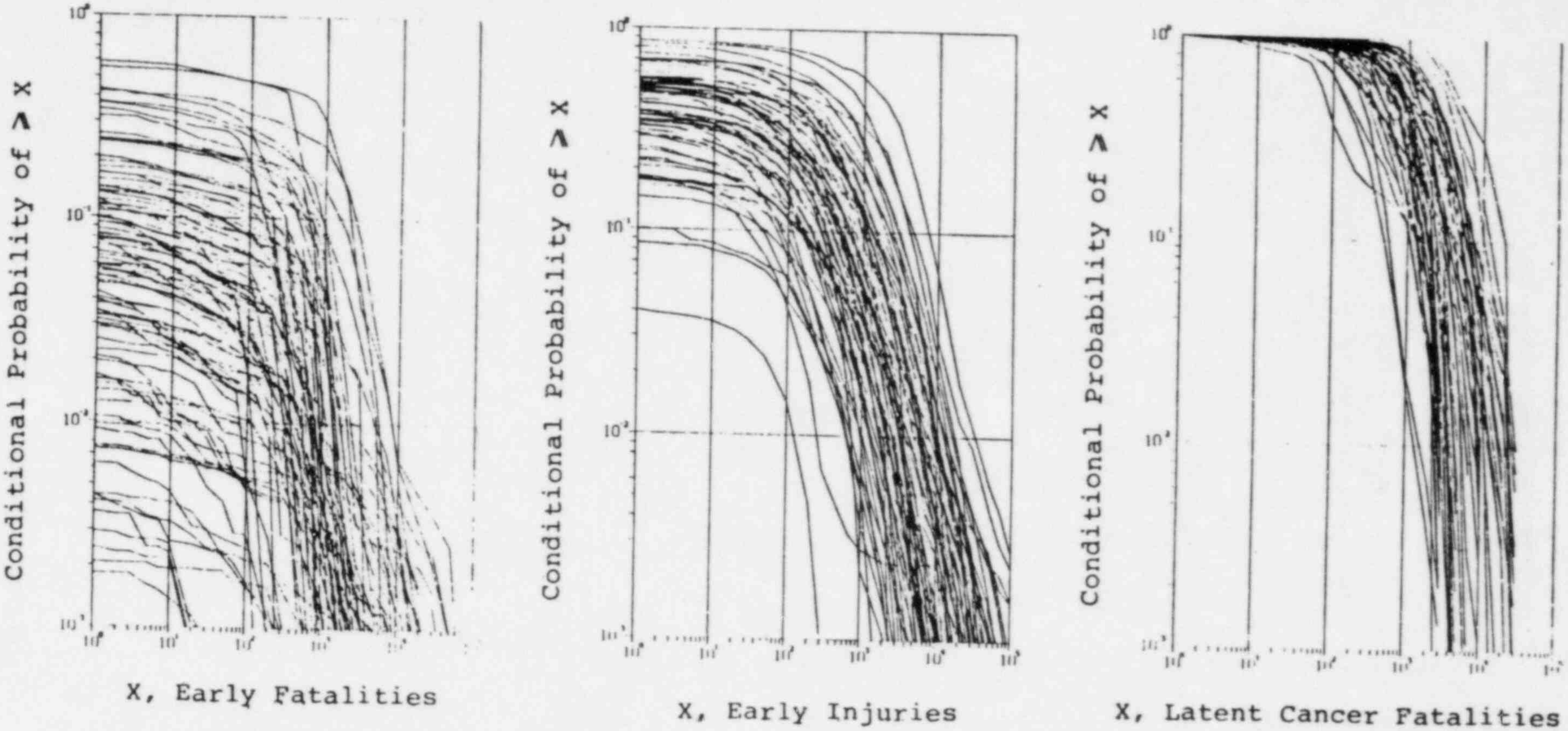


Figure 2.4.2-1. (a) Early Fatality, (b) Early Injury, and (c) Latent Cancer Fatality CCDFs Conditional on an SST1 Release at all 91 Current U.S. Reactor Sites. <u>Assumptions</u>: 1120 MWe reactor, Summary Evacuation, representative meteorology. Range of means: early fatalities 0.4 to 970, early injuries 4 to 3600, and latent cancer fatalities 230 to 8100.

Note: These CCDFs do not represent effects from existing reactor/site combinations, all assume an 1120 MWe reactor. In addition, these results are conditional on the occurrence of a hypothetical SST1 release. Recent evidence suggests that the source term magnitude assumed for SST1 may be overestimated by a factor of 10 or more (see section 2.3.2).

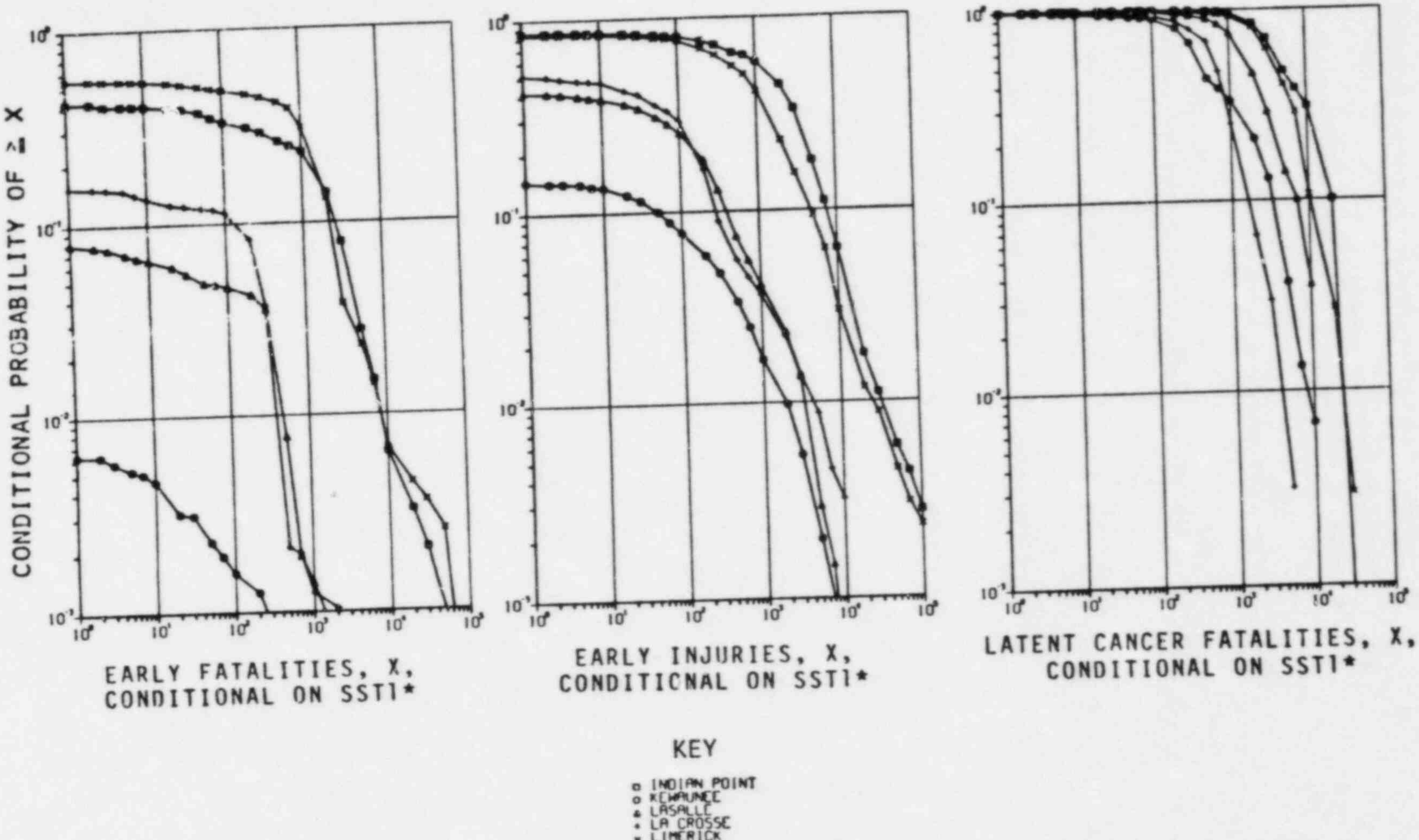


Figure C-8: Early fatality, early injury, and latent cancer fatality CCDFs at named sites, conditional on an SST1 release.
Assumptions: 1120 MWe reactor, summary evacuation, representative meteorology (see Appendix A), and actual site population and windrose.

*See footnote, page C-2.

PART B: Perspectives on the Acceptability of the Risk

1. Comparative Risk

Q1. What are the staff's practices regarding the use of comparative risk perspectives in the regulation of power reactor plants?

A1. In recent years it has been the practice of the staff to be alert to clues that a particular reactor or group of reactors may pose a disproportionate share of the risk posed by all power reactors licensed to operate by the Commission. Likewise the staff also looks for clues that a particular class of accident sequences may be dominant contributors to risk at a particular plant or group of plants. Where clues to the origin of dominant contributors to risk are found, this information has been used in the allocation of staff priorities: staff efforts are focused on reducing dominant contributors to risk among plants or among the accident sequences of individual plants.

Q2. What examples of this practice are material to this case?

A2. Prior to the issuance of the Commission's Memorandum and Order establishing this hearing, the Office of Nuclear Reactor Regulation noted that the particularly high population densities surrounding the Indian Point, Zion, and Limerick sites tend to suggest that these plants might pose a disproportionate share of the societal risk posed by all power reactors licensed by the Commission if all other factors influencing the risk were equal. NRR established a staff task action plan to investigate the risk and to investigate possible compensatory risk reduction strategies. This effort resulted in the publication of NUREG-0850 and the staff testimony on

Commission Question One provides an updated account of the results of this inquiry.

The Commission suggested a similar approach in its January 8, 1981 Memorandum and Order establishing this hearing:

> "The Commission intends to compare Indian Point to the spectrum of risks from other nuclear power plants, since the primary basis for the Commission's decision will be how extreme are the individual and societal risks associated with Indian Point compared to the spectrum of risks from other operating stations."*

The staff employed both inter-plant and intra-plant comparative risk perspectives in identifying the "fixes" that the staff the Indian Point licensees in the Fall of 1982. Concern for the risk implications of the high population density surrounding the Indian Point site led to a particularly high priority for the staff evaluation of and response to the IPPSS. The evidence in the IPPSS and the Sandia Letter Report (later finalized as NUREG/CR-2934) suggested that four classes of accident sequences were responsible for a large part of the offsite radiological risk posed by the two-unit station, so the Director of the Office of Nuclear Reactor Regulation asked appropriate members of the staff to give top priority to verifying the plausibility of the Sandia Letter Report findings and the identification of ways to reduce the vulnerability of the plants to these four dominant risk contributors. Simple and relatively inexpensive ways were found to reduce the likelihood of all four dominant accident sequences. The staff was prepared to order these fixes, if necessary and/or

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*/ op cit. p. 8

recommend them to the Board. However, orders proved to be unnecessary. Consolidated Edison was already planning to implement two of the three applicable to Unit 2, and voluntarily agreed to the third. PASNY agreed during the hearing of Commission Question One testimony to implement the fix applicable to Unit 3 prior to restart, (The four fixes were described in staff testimony on Commission Question One and will be discussed in Part C of this testimony below.)

Q3. What inferences can be drawn for regulatory decision-making from the comparative assessments of individual risks, considering the Directors Order of February 11, 1980 (Commission Question 2) and considering emergency preparedness, as assessed in Part IV B of the staff testimony on Commission Question 1?

A3. All indications suggest that the risk to individuals posed by severe reactor accidents at Indian Point Units 2 and 3 are well within the spectrum of risks to individuals posed by accidents at other nuclear power plants licensed to operate by the NRC.

The population density surrounding the Indian Point site has virtually no effect upon risks to individuals. One could envision that the risk to individuals might be affected by the population density via the speed of evacuation. However, the inquiry into the influence of emergency preparedness and emergency response in Section IV B of the staff testimony on Commission Question One provides a reliable basis to conclude that neither the population density nor the compliance issues surrounding the emergency preparedness requirements influence individual risks appreciably.

The evidence on the risks to individuals posed by "internal" accidents, i.e., those triggered by spontaneous LOCAs, transients, transient-induced LOCA, and loss of offsite power - but not internal fires - is quite clear. Indian Point Units 2 and 3 are among the safer plants licensed to operate by the Commission with respect to the risk to individuals posed by "internal" severe reactor accidents.

Figure 3 in the preceeding Section A of this testimony package demonstrates that the Indian Point Units are well below average in the frequency of severe releases of radiation due to the widely studied "internally-initiated" accidents. This finding is confirmed by qualitative considerations and engineering judgment as well as PRA bottom line results. The origin of this finding can be traced to 1) better than average severe accident prevention features in the original plant design,1/ 2) the effects of the Directors Orders of February 11, 1980,2/ and to 3) better than average severe accident mitigation.3/

The only other factor entering into a comparison of individual risks among different plant sites, beyond the comparative likelihood of severe releases and the comparative effectiveness of emergency response, is the weather. The Indian Point site is entirely typical in the distribution of

1/ See, e.g., staff testimony on CQ1. Sections III A, IV A and C, and Part A above in this CQ5 testimony.

2/ See staff testimony on CQ2.

3/ See, e.g., staff testimony on CQ1, Sections III B, IV A and C, and CQ5 Part A above.

weather conditions as they affect the consequences of severe reactor accidents. Thus we may conclude that the Indian Point Units pose below-average risks to individuals due to "internally-initiated" severe reactor accidents.

The risks to individuals posed by "externally initiated" severe reactor accidents is less clear, principally because we have little basis for comparison. A comparable external event PRA has been published only for Zion. It is plausible that the Indian Point units were average to above average in the likelihood of severe releases of radiation due to external initiators prior to the recent fixes inspired by the IPPSS. These fixes have substantially reduced the likelihood of severe releases due to the risk-dominant seismic, fire, and hurricane vulnerabilities of the two-unit station.[4/] It is plausible that the risks to individuals due to external initiators is now average to below average, though the absence of a good basis for comparison and the particularly large uncertainties surrounding external event PRA means that this is not a particularly reliable finding. In any case, the staff finds no reason to suspect that the risks to individuals posed by severe reactor accidents at Indian Point Units 2 and 3 lie above the spectrum of individual risks posed by other reactors licensed to operate by the Commission. This can be seen by drawing in the frequency of severe releases for Indian Point Units 2 and 3 from Table B.3 below on Figure 3 of the preceeding Section A. Even when the frequency of severe releases at Indian Point, after fix, are compared with severe release frequency from internal events only at other plants, Indian Point appears to be roughly average.

4/ See figure IV A-1 in Part IV.A of the staff testimony on CQ1.

Q4. What inferences can be drawn for regulatory decision-making from the comparative assessments of societal risks?

A4. Societal risks differ from individual risks in that the societal measures add up each of the individual risk contributions over the population at risk. Therefore, the societal risks are proportional to the population density distribution. It is well known that for radii up to 50 miles, the population surrounding Indian Point is roughly ten times that of the median site (see, e.g., NUREG-0715). The Zion and Limerick sites are quite similar. Thus the size of population at risk around Indian Point is near the top, but not well above the spectrum of populations surrounding other plants licensed to operate by the Commission.

The better-than-average individual risk posed by the widely-evaluated "internally-initiated" accidents compensates for higher-than-average number of people and higher-than-average property values surrounding the site. We can conclude, as did NUREG-0715, that for the "internally-initiated" accidents, the societal risk posed by Indian Point is roughly average; i.e., well within the spectrum of societal risks posed by other domestic power reactors.

The societal risk comparison is ambiguous when externally initiated accidents are considered. It is plausible that the high population density together with the dominant seismic, fire, and hurricane vulnerabilities of the plant led to above-average societal risks for the before-fix case, i.e., prior to the recent outages in the fall and winter of 1982. The one truely comparable PRA of another reactor plant is that of Zion. The comparison (for the before-fix case) is shown in Figure 4 of Section A testimony above. Note that the uncertainty bands for the frequency of severe releases of radiation for the two Indian Point units overlap with each other and with those for Zion. Since Zion has a similar site, one can infer that Indian Point did not - in prior years - pose a level of societal risk that was well above that of any other plant, but we cannot be sure that Indian Point did not pose a disproportionate share of the societal risk compared with all the plants licensed to operate by the Commission.

The recent fixes have substantially lowered the dominant externally-initiated contributors to the risk at Indian Point Units 2 and 3. The paucity of comparable PRA's still limits our ability to draw reliable comparisons, but it is less likely that the Indian Point Units will pose a disproportionate share of the societal risk in the future than in the past.

Note that the staff testimony on Contention 1.1 demonstrates that the societal risks posed by the Indian Point Units do not loom large - in fact are quite small - compared with the background of non-nuclear societal risks.

Note also that the staff testimony above draws upon the same risk analysis reported by the staff in its testimony on Commission Question One. Unless otherwise noted, the staff testimony here and in Commission Question One does not reflect many features of Amendment One to the IPPSS. In particular, the fixes credited by the staff do not include the alterations to the control room ceiling panels to reduce their seismic fragility. One other aspect of IPPSS revisions by the licensees may result in lowering the staff's assessment of the risk. It is the reassessment of the seismic fragility of the containments at both units. The staff review of the IPPSS Amendment One and letter on the containment fragility is not complete.

2. Safety Goals

Q5. What frames of reference to illuminate risk acceptability do you propose to employ?

A5. I will, in turn, compare the risk results reported by the staff in Commission Question 1 testimony with the the quantitative design objectives in the Commission Policy Statement "Safety Goals Development Program" of March 1982* and "Safety Goals for Nuclear Power Plants: A Discussion Paper", NUREG-0880 For Comment, February 1982, and employ new evaluation estimates to explore the limitations of the benefit/cost formulation of the safety goals.

Q6. What is the status of the efforts by the NRC to develop safety goals for nuclear power plants?

A6. The Commission has recently issued a policy statement and is preparing a revision of NUREG-0880. Perhaps the most significant difference between the 1982 version of NUREG-0880 For Comment and the March policy statement is this: the two years following the issuance of the policy statement will be an evaluation period rather than a period of trial use. The safety goals, with the possible exception of the benefit/cost guideline, are not to be the basis for regulatory decision-making during the evaluation

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*/ The safety goal policy statement can be found in 48 FR 10772 March 14, 1983.

period. Minor changes in phraseology have been made in the qualitative goals. What were described as "provisional numerical guidelines" in NUREG-0880 For Comment, are now described as "quantitative design objectives". Rephrasing is being considered for the mortality risk objectives but the current policy statement preserves the concept that the contribution to the risk of mortality, both early fatality and latent cancer fatality, evaluated separately, should not exceed one tenth of one percent of the background, non-nuclear risk. The current drafts also preserve the original concept of employing $1000 per person-rem avoided as the monitization algorithm for use in benefit/cost decision-making, although its applicability is altered: the principal application of the cost/benefit guideline will be to cases in which one of the other quantified safety goals is not met. Unlike the other quantified goals, the benefit/cost guideline is described as "for trial use" in the current policy statement. It, alone, appears to be potentially applicable to the case at hand. The phasing of the plant performance guideline is unaltered from NUREG-0880 For Comment.

Q7. Please elaborate upon the applicability of the quantitative design objectives in the recent safety goal policy statement to the case at hand.

A7. Although the safety goal policy statement makes it clear that the goals are not to be used by the staff to make licensing decisions, the staff feels that a comparison of the quantitative design objectives with the risk assessment results provided in the staff's Commission Question One testimony is illuminating, both for the case at hand and for the evaluation of the proposed safety goals. However, there are many diverse and redundant reasons why the quantitative design objectives are not requirements the Indian Point Units must meet, nor should they be the exclusive basis for deciding the case. Among these are:

° The policy statement indicates that the goals are for evaluation, not for use in regulatory decision-making (with the possible exception of the benefit/cost algorithm).

° The goals are intended to describe objectives toward which the evolution of reactor safety requirements will tend. They are not envisioned as candidate requirements for operating plants, nor are they intended to be thresholds of minimally acceptable risk.

° Neither the Commission nor the staff believes the current state-of-the-art in reactor risk assessment is sufficiently precise to enable measurements to be made reliably of compliance with thresholds of minimally acceptable risk - were such thresholds to be identified.

Other comments on the applicability of particular quantitative goals are included in the comparison below.

Q8. How does the first design objective, addressing individual and societal mortality risks, read in the recent policy statement?

A8. The first objective reads:

"The risk to an average individual in the vicinity of a nuclear power plant of early fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed."

Q9. How does the mathematical comparison suggested by this objective compare with that described in NUREG-0880 For Comment, February 1982?

A9. It is the same mathematical comparison.

Q10. How do the staff calculations of early fatalities expected of severe reactor accidents at Indian Point Units 2 and 3 compare with the first design objective?

A10. The frame of reference suggested in NUREG-0880 For Comment and the recent safety goal policy statement is the biologically average individual within one mile of the site. The population data employed in the staff calculations suggest that roughly 4642 people reside within 1½ miles of the reactor, i.e., within roughly 1 mile of the site boundary. The background, non-nuclear risk for the average individual in the U.S. population is a probability of 5×10^{-4} accidental death per year.* Thus, the expected average fatality rate within one mile of the site is roughly $4642 \times 5 \times 10^{-4} = 2.32$ accidental deaths per year. The first design objective suggests that no more than one tenth of one percent (10^{-3}) of this rate should be contributed by the early fatalities expected of severe reactor accidents. Thus, the figure for comparison is 2.32×10^{-3} early fatalities per site year.

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*NUREG-0880 For Comment, p. 22.

The comparison of this quantitative design objective with the risk calculations in Section III C of the staff testimony on Commission Question One are shown in Table B.1.

Table B.1

Comparison of Staff Risk Estimates for Indian Point Units 2 and 3 with the First Mortality Risk Design Objective: Early Fatalities within One Mile of the Site

Design1/	Emergency Response2/	Supportive Medical Treatment	Expected Fatalities per site year3/	Ratio Expected/Objective	Ref.4/
Before fix	Evac reloc/late reloc	Yes	1.70×10^{-3}	.73	C.29
"	Early reloc/late reloc	Yes	1.77×10^{-3}	.76	C.30
"	Evac reloc/late reloc	No	3.46×10^{-3}	1.49	C.31
"	Early reloc/late reloc	No	3.52×10^{-3}	1.52	C.32
After fix	Evac reloc/late reloc	Yes	7.46×10^{-4}	.32	C.10
"	Early reloc/late reloc	Yes	7.93×10^{-4}	.34	C.17
"	Evac reloc/late reloc	No	1.09×10^{-3}	.47	C.11
"	Early reloc/late reloc	No	1.13×10^{-3}	.49	C.18

1/ Fixes refer to 1) Unit 2 control building structural seismic modification, 2) Unit 2 interim fire fix, 3) Unit 3 interim fire fix, 4) Unit 2 anticipatory shutdown for hurricanes. All are or will be in place prior to operation in 1983.

2/ See Sections III C or IV B of staff testimony on CQ1 for definitions of emergency response assumptions. In particular see Table III.C.2 for the formal definitions.

3/ Expected early fatalities per site year include contributions from Units 2 and 3.

4/ References are to table numbers in Section III C of the staff testimony on Commission Question One. All of the societal risk estimates within one mile are counted, and half the societal risk for the annulus from 1 to 1½ mile are counted, as was done for the background, non-nuclear risk, to apprximate one mile from the site boundary.

Q11. What inferences can be drawn from the safety goal comparison in Table B.1?

A11. Note that for the "before fix" case, i.e., the staff estimates for the plants as they were designed and operated prior to the recent outages in the fall of 1982, the results straddle the design objective. The assumption of supportive medical treatment distinguishes the over-objective and under-objective cases. These are limiting cases. When supportive medical treatment is assumed, everyone needing supportive medical treatment is modeled as receiving it. When no supportive medical treatment is assumed, no one is modeled as receiving it. In reality, we expect an intermediate case, which would result in estimates very close to the safety goal.

Note also that the assumption of evacuation yields only slightly lower early fatality projections than does early relocation. This is consistent with the finding of Section IV B of the staff testimony on Commission Question One that anticipatory evacuation, as distinct from relocation from highly contaminated areas after plume passage, offers very little additional risk reduction for those accidents not triggered by non-nuclear regional disasters: earthquakes or hurricanes. Neither evacuation nor early relocation is assumed for earthquake or hurricane-triggered accidents.

The estimates for the "after fix" case, i.e., crediting most (but not all) of the alterations in plant design made in the fall and winter of 1982-1983, are consistently below the safety goal design objective for early fatalities.

The uncertainties surrounding the staff estimates of risk are larger than the margin by which the risk falls below the design objective.

We cannot be certain that the plant risks are really below the objective. On the other hand, the uncertainty discussion presented in IV C of the Commission Question 1 testimony suggests that it is quite unlikely that the excess risk of early fatality originating from severe reactor accidents at Indian Point represents more than a few tenths of one percent of the background, non-nuclear risks for those in the vicinity of the site, and is quite likely to be well below one tenth of one percent.

Q12. How does the second design objective, addressing individual and societal mortality risks of latent cancer, read in the recent policy statement?

A12. The second objective reads:

> "The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes."

Q13. How does the mathematical comparison suggested by this objective compare with that in NUREG-0880 For Comment?

A13. It is essentially the same as one of the two tests suggested for latent cancer fatalities in NUREG-0880 For Comment. The latter document suggested that a comparison should be made in the near field (one mile of the site) and in the far field (within 50 miles of the site). The current drafts of the policy statement have dropped the near field test because it is believed to be redundant; i.e., not controlling. The new formulation of the design objective has been broadened to include risks from normal operation as well as accidents, but the policy statement indicates that it is well known that normal releases contribute very little to the total of normal and accidental risks, so that it is unnecessary to add in the risks posed by normal operation into the calculation.

Q14. How do the staff calculations of latent cancer fatalities expected of severe reactor accidents at Indian Point Units 2 and 3 compare with this, the second design objective?

A14. The frame of reference is the biologically average individual within 50 miles of the site. The population data employed in the staff calculations of risk suggest that 15,480,000 people reside within 50 miles of the plant. The background cancer mortality rate is 1.9×10^{-3} cancer fatalities per person per year in the U.S.* Thus we expect roughly 29,000 cancer fatalities per year among the population within 50 miles of the site. One-tenth of one percent of this figure amounts to 29 cancer fatalities per year. Technically, this is the design objective as it applies to the Indian Point site. This is not to say that either the staff or the Commission would be comfortable with a projected casualty rate as high as 29 deaths - even latent fatality commitments taking place many years after exposure - per year of operation of the two reactors at the site.

The risk calculations performed by the staff for dose commitments which would ultimately result in cancer fatalities in the many years after exposure are shown in Table B.2.

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*NUREG-0880 For Comment, p. 23.

Table B.2

Comparison of Staff Risk Estimates for Indian Point Units 2 and 3
with the Second Mortality Risk Design
Objective: Cancer fatalities, including thyroid and
non-thyroid cancers, from early and chronic exposure
within 50 Miles of the Site

Design	Emergency Response	Expected Latent Cancer Fatalities per per site year	Ratio Expected/Objective	Ref. Table
Before fix	Evac reloc/late reloc	1.78	.06	C.33
Before fix	Early reloc/late reloc	1.78	.06	C.34
After fix	Evac reloc/late reloc	0.245	.0084	C.12
After fix	Early reloc/late reloc	0.246	.0085	C.19

Q15. What inferences can be drawn from the safety goal comparison in Table B.2?

A15. The projected risk of cancer fatalities per site year fall far below the design objective in the proposed safety goal policy statement. Although the uncertainties are not all amenable to quantification, it is clear that it is extremely unlikely that the Indian Point Units, operating together, pose anywhere near one tenth of one percent of the background risk of cancer fatalities to the average individual within 50 miles of the site.

Q16. How do the cancer fatality risk projections prepared by the staff compare with the near field provisional numerical guideline in NUREG-0880 For Comment that has been deleted from recent safety goal drafts?

A16. The comparison is summarized below:

- Background near-field cancer rate:

4642 people within one mile X 1.9 X 10^{-3} cancer fatalities/person year = 8.82 cancer fatalities/year within one mile.

- Guideline at 0.1% of background = 8.82 X 10^{-3} cancer fatalities/year within one mile.

- Indian Point risk estimates for cancer fatality dose commitments, originating within 1½ miles of the reactors, both reactors contributing, including early and chronic exposure, thyroid and non-thyroid cancers:

Before fix, evac reloc/late reloc:	2.41 E-3	(27% of guideline)
Before fix, early reloc/late reloc:	2.80 E-3	(32% of guideline)
After fix, evac reloc/late reloc:	4.27 E-4	(4.8% of guideline)
After fix, early reloc/late reloc:	5.68 E-4	(6.4% of guideline)

Q17. What inferences can be drawn from this comparison?

A17. The assessed risk of near-field cancer fatalities is well below the goal suggested in NUREG-0880 For Comment, and this is a less stringent test then is the near-field early fatality design objective, so that this case confirms that the near field cancer guideline is not controlling.

Q18. How does the third design objective, known as the benefit/cost guideline, read in the safety goal policy statement?

A18. The benefit/cost guideline reads:

> "The benefit of an incremental reduction of societal mortality risks should be compared with the associated costs on the basis of $1000 per person-rem averted."

Q19. Is this essentially the same guideline as that in NUREG-0880 For Comment?

A19. Yes.

Q20. Where will the application of this benefit/cost guideline to Indian Point appear in this testimony?

A20. The benefit/cost guideline will be applied to shutdown, past, and some possible future alterations to the Indian Point Units in Part C of this testimony below.

Q21. How does the fourth design objective, addressing plant performance, read in the safety goal policy statement?

A21. The fourth design objective reads:

> "The likelihood of a nuclear reactor accident that results in a large-scale core melt should normally be less than one in 10,000 per year of reactor operation."

Q22. How do the staff risk estimates compare with the design objective on the frequency of large scale core melt?

A22. The staff's estimates of the severe core damage frequency (neither the staff nor the licensees distinguished severe core damage from large scale core melt) are above the design objective, i.e., a frequency at or below 10^{-4} events per reactor year. For Unit 2, the staff estimates are 10^{-3} per year for the before fix case and 3.5 X 10^{-4} per year for the after fix case. For Unit 3, the staff estimate is 6.8 X 10^{-4} per year in the before fix case and also 3.5 X 10^{-4} per year in the after fix case.

There are some acknowledged conservatisms in the staff analysis. For example, we have not credited the strengthening of the ceiling panels in the control room of either unit, nor have we eliminated the conservatisms in the after fix frequency of hurricane-induced severe reactor accidents. However, we are not confident either unit has a core melt frequency below the design objective. In any case the uncertainties in the frequency estimate are substantial; all we can reliably conclude is that the assessed frequency is roughly in the high side of the neighborhood of the design objective.

Q23. What is the staff's view of the regulatory significance of the finding that the core melt frequency at Indian Point Units 2 and 3 may each be above the design objective?

A23. The staff has done a thorough study of the frequency of nine release categories and the consequences of each at Indian Point. A large part of the residual (after fix) frequency of core melt accidents fall in the very well-mitigated release categories H and I. Since these two release categories have comparatively minor offsite radiological consequences (See Table III.C.5 of the staff testimony on Commission Question One), they are primarily an issue in the risk to the utilities' investment in their facility, rather than an issue of public health and safety.

The frequency of the more severe release categories A, B, and C, which together give rise to nearly all of the offsite radiological risk, is well below 10^{-4} events per t year in the after fix case. See Table B.3 below.

Table B.3

Combined Frequency of the Severe Release Categories A, B, and C for Indian Point Units 2 and 3*

Unit	Combined frequency of Release Categories A, B, and C per unit year	
	before fix	after fix
Unit 2	3.0×10^{-4}	3.6×10^{-5}
Unit 3	1.5×10^{-4}	1.8×10^{-5}

The frequency of these three more severe release categories we believe to be a far better measure of offsite radiological risk than the overall core melt frequency.

Were the safety goals requirements that the Indian Point Units should meet (they are not) then the effect of a high core melt frequency estimate might be to activate the quest for cost-effective risk reduction strategies under the benefit/cost guideline.

Q24. Does the staff recommend the use of the benefit/cost guideline for selecting among hypothetical changes in the design and operation of Indian Point Units 2 and 3 to reduce risk?

A24. Yes, the staff does recommend the benefit/cost guideline as one of the bases, though not the exclusive basis, for decisions concerning regulatory action to further reduce the risk posed by severe reactor accidents at Indian Point. The benefit/cost guideline is described as being for trial use in both NUREG-0880 For Comment and in the recent safety goal policy statement. We also think it a suitable measure to take into consideration the high population density surrounding Indian Point. Unlike the mortality based design objectives, which are independent of population density, the total person-rem calculations employed in the benefit/cost guideline reflect the absolute number of people at risk.

Q25. Why does the staff recommend against using the benefit/cost guideline as the exclusive basis for selection among risk reduction strategies?

A25. First, the regulations remain the principal basis for setting requirements. Second, there are large uncertainties surrounding our estimates of person-rem or person-rem averted. Many risk reduction strategies may fall in the gray area in which cost-effectiveness is in doubt. Consideration of regulatory compliance, defense-in-depth, and engineering judgment properly enter into the resolution of these ambiguities. Third, a benefit/cost algorithm based exclusively on person-rem averted may give biased results. A person-rem evaluation cannot resolve among alternative risk-reduction strategies that yield the same integrated dose avoidance but which differ in early fatality avoidance, property damage avoidance, or on-site loss avoidance. These considerations, we believe, belong in an evaluation of regulatory options to reduce risk.

Q26. Can a more thorough accounting of the cost of accidents at the Indian Point reactors be given?

A26. Yes, it is possible to estimate many of the societal costs associated with the risk predictions. An attempt to make such an estimate is included as Appendix One to this testimony.

Q27. How do the incentives for risk reduction in the draft Commission safety goals compare with those obtained from the economic analysis of risk?

A27. The two ways of calculating the economic incentives for risk reduction are in remarkably good agreement overall. The benefit-cost algorithm in the safety goal suggests that it is worth $1.8 million per year to eliminate the severe accident risk from Unit 2, whereas the monetization of risks suggests that it is worth between $0.4 and $1.1 million per year to avoid off-site radiological risk, and between $3.1 and $3.7 million per year to avoid on as well as off-site losses. For Unit 3 the safety goals suggest that risk elimination is woth $1 million per year, whereas the monetization of risks in Appendix One suggests that it is worth between $0.4 million and $0.6 million per year to avoid off-site radiological risk, and between $2.9 and $3.2 million per year to avoid on as well as off-site losses. See Appendix One.

Note that the good agreement on the dollar incentive for risk reduction is somewhat fortuitous. The benefit-cost algorithm in the safety goal is based exclusively on person-rem, whereas the economic analysis of risk covers many contributions not directly proportional to or even related to off-site exposure. We should not infer that the two ways of projecting a dollar incentive agree in every application. The figure of $1000 per person-rem averted places a much higher value on the incentive to avoid low dose exposures than do the three risk monetization estimates. When these ways of estimating the value of accident prevention are applied to Indian Point, they give similar results. If they were to be applied to the incentives for severe accident mitigation or emergency response at Indian Point, they could give quite different answers.

If we had a choice of reducing expected doses by the same fraction through improved prevention, containment, or emergency response, the safety goal algorithm would assign the same dollar value for each. However, the economic analysis of risk would give quite different results for the three cases. Prevention acts on all effects of accidents. Containment works on off-site losses but not plant damage. Emergency response works only on doses to people, not off-site or on-site property damage.

If the Indian Point plants were located at a site with average population densities, then the dollar incentive for risk reduction inferred from the safety goal benefit-cost guideline would be one tenth as large as it is, since it is proportional to the population that might be exposed within 50 miles. On the other hand, the risk monitization estiamtes in the Appendix would suggest that the value of risk reduction would be only slightly diminished. Thus it is clear that the two approaches to evaluating the incentives for risk reduction are not formally equivalent.

Q28. What regulatory inferences can be drawn from the internal structure of the monetization of projected losses in Appendix One?

A28. Since on-site losses are projected to be larger than off-site losses, improvements in severe accident prevention are more likely to be cost-effective than improvements in mitigation of the effects of severe accidents.

Since property damage and delayed health effects are the largest contributors to off-site losses, and since neither are significantly influenced by off-site emergency response, improvements in off-site emergency preparedness are likely to be less influential, and thus less cost-effective, than improvements in containment effectiveness.

Appendix One to the Direct Testimony of
Frank H. Rowsome on Commission Question 5, Section B and C

Economic Evaluation of Projected Severe Accident Losses

Q1. What is the purpose of this Appendix?

A1. The purpose of this Appendix is to explore some examples of ways in which the many contributors to severe accident risk posed by the Indian Point Units 2 and 3 might be monetized, in order to illuminate the strengths and weaknesses of the safety goal algorithm for valuing risk reduction at $1000 per person-rem averted and to gain perspective on the comparative importance of the many contributors to reactor accident risk.

Q2. Does the Commission have a policy on the dollar equivalent of health effects?

A2. No, neither the Commission nor the staff has a policy or even a rule of thumb on such conversion factors, except insofar as they may be implicit in the guidance that expenditure of up to $1000 is warranted to avoid a dose of one person-rem. The ones to be proposed here are ones I have selected to illustrate what can be done with economic tradeoff studies. They are not Commission policy.

Q3. Do you propose to put a dollar value on human life for use in regulatory decision making?

A3. No, my intent is not to place a value on real deaths or injuries. A distinction can be drawn between the value of statistical risk reduction and the value of a human life. Every one of us makes numerous decisions about whether the costs of risk avoidance are worth while. An individual makes such a decision when he elects to use or not use safety belts in automobiles. A community puts an implicit value on public health and safety when it sets budgets for police or fire departments. Each decision by the NRC on the need for more stringent safety requirements puts an implicit value on risk reduction. Thus the assignment of dollar values to health risk predictions merely serves the function of laying open for inspection the value of risk reduction that is implicit in any regulatory decision.

The problem is one of resource allocation, in my view. If the NRC were to order a shutdown of the plants, or expenditures for improved safety, the costs would be passed on to the rate payers. Society will suffer if the NRC is highly inconsistent or sets too high or too low a standard of safety. In principal, there is an optiumum level of safety that provides the greatest good for the greatest number. I do not know what that optimum is, in part because of the uncertainties in the risk assessment, in part because we do not know the precise costs of safety improvements, and in part because we do not have a societal consensus on the value that should be accorded to risk reduction. Nevertheless, an economic analysis can be helpful in illuminating these issues and we can identify extreme cases of over- or under-regulation in the interest of reactor safety.

Q4. How have you gone about arriving at dollar equivalents of health effect risk reduction?

A4. The assignment of a dollar value for health effect risk reduction for use in risk-cost tradeoffs is a difficult and potentially controversial subject. One apprach is to use dollar values for risk reduction implicit is other societal decisions. One can measure how much is spent per life saved in many contexts, such as improved highway safety, community medical services, and in many occupational safety areas. Several authors have tablulated how much society does pay per life saved or casualty averted. See, e.g., "Value of a Life: What difference Does It Make?" J. D. Graham and J. W. Vaupel, Risk Analysis Vol. 1, pp. 89-95 (1981) or B. L. Cohen, "Society's Valuation of Life Saving in Radiation Protection and Other Contexts," Health Physics Vol. 39, pp. 33-51, 1980. Society appears to be inconsistent in its willingness to incur costs to lower statistical risks. The problem is further complicated by the fact that individuals place a different value on risk reduction in different contexts. In light of the ambiguities, I will consider a spectrum of values for my illustrative calculations. The figures I propose to use are shown here in Table IV.B.3.

The high value for early fatalities and cancer fatalities are drawn from the safety goal proposed by the Advisory Committee on Reactor Safeguards in NUREG-0739. The low value approximates an intermediate value from non-nuclear contexts (see references above).

Latent cancer fatalities or genetic effects seem to warrant less expensive prevention strategies than do early fatalities, principally because they are distributed over a long period after the accident. This is not meant, however, to belittle the human suffering involved. Most of the affected individuals have the opportunity to live out a large part of their natural lives. Then, too, the cancer fatalities and genetic effects entail little societal disruption. They are widely distributed in space and time, even for the severe end of the accident spectrum. They would not significantly alter the frequency of cancer fatalities or genetic effects in the affected population. The random variation in cancer and genetic defect occurrence would mask the increase due to all but the most severe reactor accidents.

Note, however, that this is the first time many of these risks have been monetized. The values are not based upon thorough research. All the cautions cited elsewhere for innovative, unproven aspects of risk assessment apply to this economic analysis. More research and public comment would clearly be in order before such risk-cost conversion factors were to become NRC policy.

The other figures in Table A1-1 reflect my judgment of reasonable incentives for risk reduction.

Table A1-1 Hypothetical Values of the Economic Incentives to Reduce Reactor Accident Health Risks

	Low	Medium	High
Early fatality (each)	$300,000	$1,000,000	$5,000,000
Latent cancer fatality	$100,000	$300,000	$1,000,000
Genetic Effect	$30,000	$100,000	$300,000
Doses requiring supportive medical treatment	$30,000	$100,000	$300,000
Early injury	$10,000	$30,000	$100,000
Interdiction of contaminated reservoirs or rivers (liquid pathways)		$1000 per person-rem projected if no interdiction were to take place	
Non-fatal cancers	$10,000	$30,000	$100,000
Non-cancerous thyroid nodules	$3,000	$10,000	$30,000
Medical screening costs		$1 per person-rem projected	

Q5. What other accident risks do you propose to consider?

A5. Costs associated with evacuation, relocation, decontamination of offsite buildings and land, and costs associated wtih loss-of-services from regions or agricultural products that must be interdicted are calculated in the CRAC analyses of reactor risks and are listed under the heading of "property damage." The liquid pathway consequences discussed by Dr. Codell in IIID of the staff testimony on Commission Question One can be included. These, too, are counted among the economic risks originating from the offsite effects of hypothetical severe reactor accidents.

There are substantial "on-site" losses to be considered as well. These costs originate in replacement power purchases after the accident, the cleanup of the plant, and of the site. These costs would be borne by the plant owners, the rate payers, the utilities' insurers, and perhaps -- in part -- by the government. I call these "onsite" losses since they originate in the damage within the site boundary. I assume that if one Indian Point unit were to incur a core melt accident, both units would be shut down indefinitely, and that the cost of replacement power would be the same as that calculated for a shutdown order, i.e., roughly a net cost of $4.3 billion, drawn from Sidney Feld's testimony to be filed on Commission Question Six.

The cleanup of Three Mile Island Unit 2 is projected to cost one billion dollars. A full core meltdown accident would probably be more expensive to clean up. I estimate that the cost might lie between one and ten billion dollars. I shall use $3 billion in the illustrative calculations.

Q6. What do you project the economic risk to be?

A6. The calculation of expected accident losses in dollar equivalents are shown for Indian Point Unit 2 in Table A1-2 and -3, and for Unit 3 in Table A1-4 and -5.

These values employ the risk analysis results obtained for the evac reloc/late reloc emergency response model.

A1-2 Monetized Risk of Severe Reactor Accidents at Indian Point Unit 2

After fix design, evac reloc/late reloc emergency response

Consequence	Average consequence per reactor year[1/]	Expected loss in dollars per year cost conversion factor		
		Low	Medium	High
Offsite Risks				
1. Early fatalties	1.48(-2)	$4,440	$14,800	$74,000
2. Latent cancer fatalities	2.09(-1)	$20,900	$62,700	$209,000
3. Genetic effects	6.78(-1)	$20,340	$67,800	$203,000
4. High Individual doses*	3.60(-2)	$1,080	$3,600	$10,800
5. Early injury	1.15(-1)	$1,150	$3,450	$11,500
6. Non-fatal cancers	4.29(-1)	$4,300	$12,900	$43,000
7. Non-cancerous thyroid	1.43(0)	$4,300	$12,900	$43,000
8. Medical screening costs		$2,610	$2,610	$2.610
Subtotal: health effects/yr		$59,000	$181,000	$597,000
9. Property damage		$281,000	$281,000	$281,000
10. Liquid pathway interdiction	via fallout	$194,000	$194,000	$194,000
	via basemat meltthrough	$53,000	$53,000	$53,000
Subtotal: offsite cleanup costs		$528,000	$528,000	$528,000
Total offsite radiological costs/yr		$587,000	$709,000	$1,125,000
Onsite Risks				
1. Replacement power		$1,510,000	$1,510,000	$1,510,000
2. Onsite cleanup		$1,054,000	$1,054,000	$1,054,000
Total onsite costs/yr		$2,564,000	$2,564,000	$2,564,000
Total of offsite and onsite losses/yr		$3,151,000	$3,273,000	$3,689,000

*Doses that would result in early fatalities in the absence of supportive medical treatment.

[1]Source: Table IIIC6, staff testimony on CQ1.

The corresponding value of risk elimination based upon the safety goal benefit algorithm is $1,790,000 per year, based upon Table IIIC8.

A1-3 Monetized Risk of Severe Reactor Accidents at Indian Point Unit 2

Before fix design, evac reloc/late reloc emergency response

Consequence	Average consequence per reactor year[1/]	Expected loss in dollars per year cost conversion factor: Low	Medium	High
Offsite Risks				
1. Early fatalties	6.99(-2)	$21,000	$70,000	$350,000
2. Latent cancer fatalities	1.57(0)	$157,000	$471,000	$1,570,000
3. Genetic effects	5.07(0)	$152,000	$507,000	$1,521,000
4. High Individual doses*	2.05(-1)	$6,150	$20,500	$61,500
5. Early injury	8.32(-1)	$8,320	$25,000	$83,200
6. Non-fatal cancers	3.16(0)	$31,600	$94,800	$316,000
7. Non-cancerous thyroid	1.05(+1)	$31,500	$105,000	$315,000
8. Medical screening costs		$19,500	$19,500	$19,500
Subtotal: health effects/yr		$427,000	$1,313,000	$4,236,000
9. Property damage		$2,100,000	$2,100,000	$2,100,000
10. Liquid pathway interdiction	via fallout	$1,630,000	$1,630,000	$1,630,000
	via basemat meltthrough	$71,000	$71,000	$71,000
Subtotal: offsite cleanup costs/yr		$3,801,000	$3,801,000	$3,801,000
Total offsite radiological costs/yr		$4,228,000	$5,114,000	$8,037,000
Onsite Risks				
1. Replacement power		$4,300,000	$4,300,000	$4,300,000
2. Onsite cleanup		$3,000,000	$3,000,000	$3,090,000
Total onsite costs/yr		$7,300,000	$7,300,000	$7,300,000
Total of offsite and onsite losses/yr		$11,528,000	$12,414,000	$15,337,000

*Doses that would result in early fatalities in the absence of supportive medical treatment.

[1]Source: Table IIIC23, staff testimony on CQ1.

The corresponding value of risk elimination based upon the safety goal benefit algorithm is $13,200,000 per year, based upon Table IIIC23.

A1-4 Monetized Risk of Severe Reactor Accidents at Indian Point Unit 3

After fix design, evac reloc/late reloc emergency response

Consequence	Average consequence per reactor year[1]	Expected loss in dollars per year cost conversion factor Low	Medium	High
Offsite Risks				
1. Early fatalties	3.75(-3)	$1,100	$3,800	$18,800
2. Latent cancer fatalities	1.14(-1)	$11,100	$34,100	$114,000
3. Genetic effects	3.72(-1)	$11,160	$37,200	$111,600
4. High Individual doses*	1.11(-2)	$333	$1,110	$3,330
5. Early injury	4.09(-2)	$409	$1,230	$4,090
6. Non-fatal cancers	2.36(-1)	$2,360	$7,080	$23,600
7. Non-cancerous thyroid	7.87(-1)	$2,360	$7,870	$23,600
8. Medical screening costs		$1,430	$1,430	$1,430
Subtotal: health effects/yr		$30,000	$94,000	$300,000
9. Property damage		$165,000	$165,000	$165,000
10. Liquid pathway interdiction	via fallout	$97,000	$97,000	$97,000
	via basemat meltthrough	$72,000	$72,000	$72,000
Subtotal: offsite cleanup costs		$334,000	$334,000	$334,000
Total offsite radiological costs/yr		$364,000	$427,000	$634,000
Onsite Risks				
1. Replacement power		$1,510,000	$1,510,000	$1,510,000
2. Onsite cleanup		$1,060,000	$1,060,000	$1,060,000
Total onsite costs/yr		$2,570,000	$2,570,000	$2,570,000
Total of offsite and onsite losses/yr		$2,934,000	$2,997,000	$3,204,000

*Doses that would result in early fatalities in the absence of supportive medical treatment.

[1] Source: Table IIIC7, staff testimony on CQ1.

The corresponding value of risk elimination based upon the safety goal benefit algorithm is $988,000 per year, based upon Table IIIC9.

A1-5 Monetized Risk of Severe Reactor Accidents at Indian Point Unit 3

Before fix design, evac reloc/late reloc emergency response

Consequence	Average consequence per reactor year[1]	Expected loss in dollars per year cost conversion factor Low	Medium	High
Offsite Risks				
1. Early fataties	3.83(-3)	$1,150	$3,830	$19,150
2. Latent cancer fatalities	8.03(-1)	$80,300	$241,000	$803,000
3. Genetic effects	2.58(0)	$77,400	$258,000	$774,000
4. High Individual doses*	2.06(-2)	$618	$2,060	$6,180
5. Early injury	2.44(-1)	$2,440	$7,320	$24,400
6. Non-fatal cancers	1.66(0)	$16,600	$49,800	$166,000
7. Non-cancerous thyroid	5.53(0)	$16,590	$55,300	$165,900
8. Medical screening costs		$9,930	$9,930	$9,930
Subtotal: health effects/yr		$205,000	$627,000	$1,969,000
9. Property damage		$1,180,000	$1,180,000	$1,180,000
10. Liquid pathway interdiction	via fallout	$836,000	$836,000	$836,000
	via basemat meltthrough	$81,000	$81,000	$81,000
Subtotal: offsite cleanup costs		$2,097,000	$2,097,000	$2,097,000
Total offsite radiological costs/yr		$2,302,000	$2,724,000	$4,066,000
Onsite Risks				
1. Replacement power		$2,906,000	$2,906,000	$2,906,000
2. Onsite cleanup		$2,028,000	$2,028,000	$2,028,000
Total onsite costs/yr		$4,934,000	$4,934,000	$4,934,000
Total of offsite and onsite losses/yr		$7,236,000	$7,658,000	$9,000,000

*Doses that would result in early fatalities in the absence of supportive medical treatment.

[1] Source: Table IIIC25, staff testimony on CQ1.

The corresponding value of risk elimination based upon the safety goal benefit algorithm is $6,590,000 per year, based upon Table IIIC27.

C. Evaluation of Possible Regulatory Actions

Q.1. What is the purpose of this section?

A.1. The purpose of Part C: Evaluation of Possible Regulatory Actions, is to provide staff perspectives on the question of whether the Indian Point Units 2 and 3 plants should be shutdown or other actions taken. The perspectives provided herein are based only upon the staff inquiry into risk, i.e., to Commission Questions 1, 2, and 5.

1. <u>Bases for Regulatory Initiatives</u>

Q.2. What are the bases the staff proposes to use in evaluating hypothetical regulatory requirements to reduce risk at Indian Point Units 2 and 3?

A.2. The primary basis for regulatory requirements applicable to Indian Point has been and will continue to be the regulations. Because of the comparatively high population density surrounding the site, as well as this hearing, the staff has been employing risk assessment perspectives to examine the need for more stringent requirements, or an altered focus for some of the requirements for Indian Point.

Probabilistic risk assessment is a very powerful tool with which to develop perspectives on the safety profile of a plant. Many insights into the strengths and weaknesses of the safety functions of a power reactor can and have been obtained. In addition, the PRA model can function as an evaluation tool for hypothetical safety issues or candidate concepts for alterations in plant design and operation intended to reduce risk. However, PRA's suffer from problems of imprecision and possible biases or omissions. The staff does not recommend taking inferences from any reactor risk assessment, including those of Indian Point, as revealed truth. Rather, inferences for regulatory action drawn from PRA's should be taken as hypothesis to be tested against the full weight of the evidence concerning the relevant safety issues. Whenever PRA-based insights are used to give shape or focus to regulatory action it is wise to entertain the hypothesis that the particular PRA insights may be wrong. In particular, we should identify and explore the assumptions in the PRA to which the relevant insights are particularly sensitive. In short, it is generally desirable to make a context-specific assessment of PRA uncertainties.

PRA-based insights have been and - in the view of the staff - should continue to be a source of perspective with which to help set regulatory priorities and to sharpen the focus of regulatory implementation, not only to better assure public health and safety but also to avoid over-regulation. Regulatory decision-making should weigh compliance with the regulations, PRA-based insights, the effects of the decision on defense in depth and, for departures from the current requirements, the benefit/cost relationship of the options under consideration. Many examples will be discussed in the following testimony.

2. Fixes Inspired by the Indian Point Probabilistic Safety Study

Q.3. What changes in plant design or operation have been credited by the staff in the "after fix" assessment of risk?

A.3. Four fixes have been credited by the staff in the "after fix" cases evaluated in staff testimony on Commission Questions One and Five above. These are:

1. Alterations to the Unit 2 control building to reduce the seismic fragility of the structure associated with interactions between the Unit 2 control building and the Unit 1 superheater building during earthquakes. This is known as the "bumper" fix. It has been implemented.

2. Alternate cables and breakers have been provided at Unit 2 to enable a component cooling water pump and a charging pump to be energized in the event that a fire in the cable tunnel or switchgear room disables the normal power supply to these systems. This is known as the Unit 2 interim fire fix. It has been implemented.

3. A fire fix for Unit 3 similar to that in Unit 2 has been proposed. PASNY has committed to installing it prior to restart.

4. Consolidated Edison has amended the Unit 2 technical specifications to mandate an anticipatory shutdown of Unit 2 should a hurricane threaten the site.

Q.4. What considerations entered into the staff decision to recommend such fixes?

A.4. The Sandia Letter Report (later finalized as NUREG/CR-2934) identified four accident sequences as dominating the risk from the two-unit Indian Point site. All four entailed loss of control or power to all motor-driven engineered safety features (damage state E), leading to the severe release category C in 40% of the occurrences, according to the staff containment analysis. One of the accident sequences originated in earthquake-induced damage to the Unit 2 control building. Two others originated in postulated fires in the cable tunnel or switchgear room of each of the two units. The fourth originated in hurricane damage to the Unit 2 control building and/or offsite power and the Unit 2 diesel generator building.

Either prevention fixes - alterations to reduce the vulnerability of the plants to the initiating events, or mitigation fixes to avoid the late overpressurization failure mode of containment could have reduced the offsite radiological risk posed by these accident sequences.

Considerations entering into the choice of prevention vs. mitigation were these:

° Speed of implementation

Consolidated Edison had already designed and was preparing to implement prevention fixes for two of the three dominant contributors applicable to Unit 2. Engineering feasibility and ease of implementation clearly favored the initiator-specific prevention fixes.

° Regulatory compliance

The fire fixes constitute partial compliance with the requirements of Appendix R. The other two prevention fixes are entirely within the spirit, if not required by the letter of the applicable regulations.

° Risk limitation effectiveness

Staff members independently confirmed the licensees' judgment that the fixes would be effective in substantially reducing the likelihood of the four dominant accident sequences. Mitigation might theoretically have proven to be equally effective, not only for these four sequences but also for the full array of damage state E accidents. No significant attendant risks were identified for the prevention fixes.

° Benefit/cost considerations

Early in the dialog between the staff and the licensee on these issues, it became clear that very economical preventive fixes could be identified for all four dominant risk contributers.

Preliminary indications suggested an overwhelmingly favorable benefit/cost relationship for the prevention conceptions. This has not been the case in staff mitigation stuc.es. Then, too, prevention helps protect the licensees' investment in the facility as well as public health and safety. As noted above, on-site losses that are unaffected by mitigation improvements contribute a large part of the economic risk profile of the station.

° Defense-in-depth considerations

Both prevention and mitigation fix options strengthen defense-in-depth, because the vulnerabilities underlying the dominant contributers are to common-cause failures that threaten both core cooling systems and containment heat removal systems, i.e., two levels of defense-in-depth.

On balance, the staff thought it reasonable and prudent to press the licensees to implement the prevention fixes.

Q.5. What are the staff's current estimates of the monetized benefit of the four prevention fixes?

A.5. One basis to estimate the monetized benefit of the fixes is to draw upon the benefit/cost guidance in the safety goal policy statement, i.e., $1000 per person-rem averted. NUREG-0880 suggests limiting the exposure evaluation to within 50 miles of the reactor site.

For the evac reloc/late reloc emergency response model the expected person rem per reactor vear can be found in Dr. Acharya's III C testimony on Commission Question One:

Unit 2, before fix:	13,200	to 50 miles
Unit 2, after fix:	1,790	to 50 miles
difference	11,410	to 50 miles
Unit 3, before fix:	6,590	to 50 miles
Unit 3, after fix:	988	to 50 miles
difference	5,602	

The expectation value of the person-rem averted in each year after the fix is thus roughly 17,000 evaluated to 50 miles. Thus the monetized value of the fix is $17 million per year. Opinions differ on the discount rate to employ in translating the annual value of risk reduction into a present worth. However, the uncertainties in the expected person-rem calculations loom large compared with the effect of the choice of discount rate, so that the choice does not have a significant influence on the range of the results. In contexts such as the present worth of the costs of shutdown, the staff commonly employs a 5% per annum discount rate. For consistency, I shall employ a 5% discount rate here. This yields a conversion factor of 15 years to convert annual benefits to a present worth.

The value of the fixes is thus roughly $255 million according to the safety goal guidance on assessing the value of risk reduction.

Another estimate of the value of the fixes can be drawn from the economic estimate suggested in Appendix One of this testimony. These suggest that the present worth of the fixes lie between $190

Million and $262 Million, in excellent agreement with the safety goal result. It may be of interest to display the breakdown of this assessment by plant and contributor. This is shown in Table C.1.

Table C.1

Value of the Four Recent Fixes at Indian Point
Units 2 and 3, Based Upon the Economic Monetization Estimates.[1/]

Unit	Case	Value of Annual Expected Losses (Millions of Dollars)[2/]				
		Health Effects			Offsite	Onsite
		Lo	Med	Hi	Property Damage	Property Damage
IP2	before fix	0.43	1.31	4.24	3.80	7.30
IP2	after fix	0.06	0.18	0.60	0.53	2.56
IP2	difference	0.37	1.13	3.64	3.37	4.74
IP3	before fix	0.21	0.63	1.97	2.10	4.93
IP3	after fix	0.03	.09	0.30	0.33	2.57
IP3	difference	0.18	0.54	1.67	1.77	2.46
Total	difference	0.55	1.67	5.31	5.14	7.20
Present worth[3/] of fixes (Millions of Dollars)		8	25	80	77	110

1/ Evac reloc/late reloc emergency response model. See also Appendix One.

2/ Based on Table A.2 through A.5 in Appendix One.

3/ Present worth = value/year X 15 years @ 5% discount rate.

Q.6. What is known about the cost effectiveness of the fixes?

A.6. The staff has not solicited a cost estimate from the licensees for the fixes, but it is thought to have been of the order of a few million dollars or less. The anticipatory shutdown of Unit 2 for hurricanes may entail a replacement power cost of the order of one million dollars should a hurricane threaten the site, which is expected at a frequency of the order of once in one hundred years. Thus the expected losses are very roughly $10,000 per year, or $150,000 in present worth. This is not a large contributor.

For a two to three million dollar estimate of the cost of the fixes, we find a benefit/cost ratio of roughly 100:1.

If all the costs of the risk-based inquiry into the safety of Indian Point were totalled, including NRC costs and licensee costs, we expect the result would fall in the range of 10 to 30 million dollars. If the only value of this entire enterprise were the four fixes, we would still find a highly favorable benefit/cost ratio of roughty 10:1.

3. Shutdown

Q.7. Might a permanent shutdown be a cost-effective risk avoidance strategy according to either the economic or safety goal algorithms for valuing risk reduction, in comparison with the costs of shutdown?

A.7. A permanent shutdown is not a cost-effective risk avoidance strategy compared with the costs of shutdown. Estimates of the costs of shutdown range from $3.5 billion (UCS) to $17 billion (Rand Corporation for the Power Authority of the State of New York). The staff estimate of the net cost of site closure is expected to be roughly $4.3 billion. See also staff testimony on Commission Question 6. The present worth of the risk averted is estimated to be (in millions of dollars):

Unit	Safety Goal	Low	Medium	High Economic Estimates
2	$26.9 M	$47.3 M	$49.1 M	$ 55.5 M
3	$14.8 M	$44.0 M	$45.0 M	$ 48.1 M
Total	$41.7 M	$91.3 M	$94.1 M	$103.4 M

Using the $4.3 billion estimate for the cost of shutdown, the benefit/cost ratios range from 1:42 to 1:100, against shutdown. Although there are large uncertainties in the staff estimate of risk* it is quite unlikely that shutdown might be a cost-effective risk avoidance strategy, according to the staff estimate of risk and the several ways of monitizing risk reduction.

Q.8. Do the values of shutdown tabulated above place an upper limit on the value of hypothetical fixes to further reduce the risk?

A.8. Yes, under the assumption that the staff assessment of risk is correct.

Since the present worth of the elimination of severe accident risk at Unit 2 amounts to $26.9 million dollars (safety goal) or 47 to

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* See Section IV C of the staff testimony on Commission Question One.

55 million dollars (economic estimates) no one-time expenditure to lower the risk posed by Unit 2 costing more than this could be found to be cost effective, unless information indicative of higher risk than that identified in the staff testimony on Commission Question One were to surface.

Note also that a large part of the value of risk reduction in the economic estimates originates in averted replacement power and on-site cleanup costs. The safety goal algorithm provides a consistently conservative evaluation of averting offsite radiological risks. Public health and safety considerations might warrant an expenditure of up to but no more than roughly $27 million on Unit 2 for the reduction of the residual (after fix) offsite radiological risk, identified in the staff Commission Question One testimony, if such residual risk reduction is to be cost-effective.

Likewise $14.8 million constitutes an upper bound on what might be cost-effectively spent on reducing the offsite radiological risk as estimated by the staff at the Indian Point Unit 3. If the reduction in the licensee's economic risk is also considered, the upper bound is roughly $48 million.

4. Further Changes in Plant Design

Q.9. Does the staff recommend that further changes in the design of Indian Point Units 2 or 3 be ordered of the licensee, beyond those already implemented, committed, or required by the regulations, on the basis of severe accident risk considerations?

A.9 . No, the staff does not recommend that the NRC order further changes in design at Indian Point Unit 2 or 3 based upon severe accident risk considerations. The staff has considered the prospects for the cost-effective reductions in the residual risk through additional design changes to improve severe accident prevention or mitigation. This analysis is described in Appendix 2 of this testimony. This inquiry turned up no fixes that are unambiguously cost effective. The prospects that additional study might turn up modest-benefit, low-cost design improvements that are cost-effective is good. Nevertheless the staff recommends against a high-priority, plant specific program to search out and require such minor improvements for the following reasons:

1) The residual risk is a small (See, e.g., staff testimony on Contention 1.1, Parts IIIC and IVA of Commission Question 1 testimony, and the safety goal considerations in Commission Question 5 Part B above.)

2) The societal risk does not appear to be well above the spectrum of risks posed by other plants licensed to operate by the Commission. (See Parts A and B of the Commission Question 5 testimony above.)

3) Other ongoing or recommended programs can suffice to harvest further insights from the inquiry into risk posed by Indian Point Units 2 and 3. The Severe Accident Policy Development

program, described to the Commission in SECY 82-1B, is a generic effort to identify whether, and if so how, the regulations need be altered to assure that severe accident considerations are adequately dealt with. This, and related generic programs dealing with unresolved or generic safety issues provide an appropriate venue for pursuing the potentially cost effective options for the reduction in the low residual risk at Indian Point Units 2 and 3.

In addition, the Safety Assurance Program we recommended in Section C.5 of this testimony below would provide another venue for the licensees to explore further risk reduction.

Q.10. To close out Dr. Meyer's discussion of mitigation in Contention 2.2.1 and Part IIIB of the staff testimony on Commission Question One, please summarize the staff position on requiring backfits to Indian Point Units 2 and 3 to better mitigate severe reactor accidents.

A.10. The staff does not recommend that the NRC issue orders to implement backfits to improve the mitiation of severe reactor accidents.

The staff, in the task action plan developed in 1980 to study the need for retrofits at Indian Point and Zion, placed a heavy emphasis on mitigation factors such as hydrogen control, controlled filtered venting of containment, and a core retention device because it was not clear, at the time, that the plants as built provided effective mitigation of the offsite radiological consequences of core melt accidents. At the time, it was plausible that hydrogen burns or pressure surges associated with vessel melt through might breach containment in most core melt accident scenarios. Our studies have shown that this is not the case. In 1980 it was plausible that the gasses evolved from core concrete interactions, together with other contributors in the pressures and temperatures in the containment following core melt might lead to gross, early overpressure failure of containment and high projections for early fatalities offsite. Our studies indicate that this is not the case. Rather, we find that gradual overpressure failure of containment will not take place with an operating containment heat removal system, will take a long time to develop in any case, and almost never result in early fatalities even in the absence of evacuation. Basement melt through has been found to produce very little offsite radiological risk. Thus most of the desirable attributes of the early mitigation backfit conceptions are already present in the plants.

The five elements that enter the decision-making process of whether or not to recommend a mitigation feature or stragegy (as outlined in the staff testimony on contentions 2.1a and d), namely risk reduction, undue risk, feature feasibility, cost, and tradeoffs between prevention and mitigation have been addressed in the staff's testimony on Commission Questions 1 (Part III.B) and in Appendix 2 to this testimony. The major element and sufficient condition for making no recommendations at this time is the staff determination that Indian Point poses no undue risk. In addition, as described above, only relatively inexpensive features (or strategies) in the range of 1 to 10 million dollars would be potentially attractive for further risk redunction under the benefit-cost guidelines presented here. Further, within this "benefit-cost" range, prevention fixes may still be attractive alternatives to mitigation. Nevertheless, certain important points presented by the staff at this Hearing should not be lost in considering mitigation strategies in general.

- They can offer additional risk reduction (e.g., 80% reduction in latent effects) by considerably extending the performance capability of the containment under core melt conditions.
- Even though a number of the more attractive mitigation features had to be eliminated due to high cost estimates (e.g., filtered vents or passive containment heat removal), there are feasible candidates in the "modest" cost range (e.g., independent auxiliary sprays).

Because of the above, the staff intends to continue pursuing "mitigation" options for operating reactors within the context of SECY 82-1B, "Proposed Commission Policy Statement On Severe Accidents And Related Views On Nuclear Reactor Regulation." This is consistent with the staff approach for Indian Point action as described on Figure 1.1 of NUREG-0850, namely that if the staff determined that Indian Point did not pose undue risk, then the matters relating to mitigation would be folded into the generic activities for all operating reactors (as described in SECY 82-1B).

5. <u>Changes in the Conduct of Operations</u>

Q.11. Does the staff recommend any orders be issued by the Commission applicable to the Indian Point licensees concerning the conduct of operations at the plants?

A.11. Yes. The staff recommends that the Commission order both licensees to implement a Safety Assurance Program with the objectives of assuring that the conduct of operations and future clues to the safety of the plants are and remain consistent with a level of severe accident risk not applicably greater than that assessed by the staff.

Q.12. What are the bases for this staff recommendation?

A.12. First, we have observed that many of the more important precursors to severe reactor accidents occurring at other plants entailed maintenance error, surveillance error, operator error, or management oversight. Virtually every historical instance in which whole redundant safety systems have been found to be inoperable can be traced, in plant, to such errors. A common element of all these occurrences was a failure by the operations personnel to fully appreciate the importance-to-risk of their own actions or of the systems or phenomena entailed. Procedures and operations staff training altered to reflect the insights obtainable from the IPPSS and staff analyses of risk could go a long way to make such errors very much less likely.

Second, staff testimony has indicated that the IPPSS credits operator actions different from or extending beyond that in current emergency procedures and/or operator training.* Where the IPPSS suggests operator response tactics that result in lower risk than adherance to the written procedures, the procedures should be re-examined and improved, if possible.

Third, the prospects for cost-effectiveness of a safety assurance program are excellent. Prior searches to cost-effective risk reduction retrofits have concentrated upon design features. This resulted in some very highly cost-effective fixes. Although the residual risk is not large, the comparatively low cost associated with studies and alteration to procedures or training suggest that such a program to maintain and harvest the insights of the PRA's for the conduct of operations are very likely to be cost-effective. We anticipate program startup costs of a few million dollars and continuing costs of a few hundred thousand dollars per year. The program would substantially lower the likelihood that the risk might be significantly higher than our best estimates suggest, or might grow to be higher in the future. Reduction in the risk to the licensees' investment in the facility (several million per unit year by our estimates) would probably equal or exceed the costs.

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* See NUREG/CR-2934, p. 2.5-1 and references therein.

Fourth, staff and UCS witnesses have pointed out that the PRA's contain few allowances for wearout. Future changes in the frequency of component failures, human errors, initiating events, precursors at other plants, or information from reactor safety research might turn up clues to higher risk. A followup program to maintain and improve the IPPSS can provide a mechanism to better assure that the lessons from such experience is thoroughly understood and, where necessary, acted upon.

Q.13. What features does the staff wish to see in the recommended Safety Assurance Program?

A.13. In broad outline, the staff believes that the Safety Assurance Program should entail:

1. Review, and when warranted, revision of procedures for maintenance, surveillance testing, operations, technical specifications, and personnel training to harvest the insights that can be obtained from the PRA's for better conduct of operations.

2. The use of the PRA's as an evaluation tool to identify the importance to risk of patterns in failure data obtained at Indian Point and to evaluate the relevance to Indian Point of severe accident precursors at other plants.

3. Continued maintenance and use of the IPPSS as an operations management and design evaluation tool, including the implementation of cost effective risk reduction concepts.

4. Integration of the Safety Assurance Program into the conduct of operations.

The staff believes that a large part of the value of the Safety Assurance Program conception lies in the familiarity gained by operations personnel on the importance to risk of their resonsibilities. To achieve this goal, it is important that the licensees integrate the program into their operations organization, and minimize the extent to which it is an external or contracted function. The staff feels it to be highly desirable that the program be home grown by the licensees rather than prescriptively imposed upon them. The staff will be open to compromise on the specifics in order to achieve the thorough integration into plant operations we recommend.

Some technical elements of the program we recommend are the following:

1. Formal calculations of quantitative measures of importance-to-risk for initiating events, systems, components, human interactions in maintenance, surveillance, and operations.

   Such figures of merit hearing upon the importance to safety can be illuminating in several ways: a) they may reveal limitations in the PRA's, b) they are useful in

the training of operators and maintenance personnel,
c) they are useful in the evaluation of procedures, technical specifications, and situations that may arise in plant operations.

2. Fault Hazards Analysis applied to hypothetical errors in the conduct of maintenance procedures, surveillance procedures, normal and emergency operating procedures, and technical specifications. This constitutes a formal "what if" examination of potential human error in the conduct of operations.

3. Where the importance-to-risk and the fault hazards analysis suggest that procedures may warrant improvement, the analysis should be extended to human error Failure Mode Effects Analysis and changes, where plausibly cost effective, should be instituted in procedures, technical specifications,
operator training, system design, and/or control room simul-tator design as appropriate.

4. Operations and maintenance personnel should be trained on the results of the studies into the importance-to-risk of their responsibilities, taught pattern recognition for the more vulnerable plant configurations, or circumstances and diagnosis of the more important accident scenarios.

5. From time to time the PRA quantification should be updated to reflect accumulated experience on the frequency of component failures, human errors, and initiating events. This effort can be made economical by employing the quantitative measures of importance-to-risk to assess the significance of altered fault event frequency, so that comprehensive and burdensom recalculations of risk are rarely necessary.

6. The licensees, with the advise and consent of the staff, should devise and implement criteria spelling out thresholds for corrective action and of reportage to the NRC of discoveries of less than expected system reliability, precedural adequacy, or greater-than-expected risk, where the IPPSS serves as the frame of reference.

7. The IPPSS models should be employed as a test bed to assess the importance to risk of events at other plants that meet the criteria to be considered precursors to severe reactor accidents in the sense of NUREG/CR-2497, which might potentially be relevant to Indian Point.

8. The results of the importance-to-risk evaluations should be made available to the licensees' quality assurance organization, NRR, and IE, not only to enable reviews to be made of its adequacy, but potentially for use in sharpening the focus or allocation of emphasis in the work of the QA and NRC audits.

9. The IPPSS and the assessments of importance-to-risk is based upon it should be maintained, and, when appropriate, revised to made it a current, up-to-date evaluation tool.

10. The Power Authority of the State of new York has underway a study of systems interactions at Indian Point Unit 3. As noted in prior testimony, this effort has been kept separate from the PRA endeavor, with the concurrence of the staff. Nonetheless it may provide valuable insights on the strengths and weaknesses of the IPPSS in this important area. The staff would like to see the IPPSS altered to reflect the effect of identified systems interactions at Unit 3 before as well as after credit is taken for any alterations in plant design or operation triggered by the systems interaction study. This before and after fix recalculation of the risk will provide an important benchmark that will help to determine whether such a systems interaction study may be needed for Indian Point Unit 2 and in many other applications of reactor risk assessments.

Q.14. What resources do you expect would be required of the licensees to implement such a program?

A.14. The program entails large startup costs. I judge that the initial work to get the program underway might entail one to three million dollars, roughly equally divided between the plant operations personnel time, home office engineering time, and contractors time.

I judge that once it is in place, one to three hundred thousand dollars per year in operations staff salaries and overhead would sustain it.

Q.15. How does the staff suggest that the Commission formulate the order?

A.15. The order should not be highly prescriptive, in order that the licensees have the freedom to develop a program well-integrated into their plant operations staff. We suggest that the licensees be ordered to develop and implement a safety assurance program, subject to the advice and consent of the staff, with the objection of assuring that the conduct of operations as well as future evidence bearing upon the safety of the plant are and remain consistent with a level of severe accident risk not appreciably greater than that assessed by the staff in its testimony at this hearing.

6. Conclusions

Q.16. Please summarize the staff risk-based conclusions.

A.16. The staff concludes that the Indian Point Units 2 and 3 do not pose undue risk to public health and safety. The staff has identified no futher changes in the design of either Unit that warrants implementation by NRC order.

The Staff does recommend ordering that a follow-up program be instituted by the licensees to harvest the insights available from the Indian Point risk assessments for improvements in the conduct of operations.

Q.17. Does this concude your testimony?

A.17. Yes, but for the Appendix.

Appendix 2 to "Direct Testimony of
Frank H. Rowsome on Commission Question 5, Parts B and C

Evaluation of the monetized benefits of further Design Changes
to Lower Risk at Indian Point Units 2 and 3

Q.18. What is the purpose of this Appendix?

A.18. The purpose of the Appendix is to describe the staff assessment of the prospects for further cost effective risk reduction at Indian Point Units 2 and 3 that might be obtained by changes in plant design. Each release category is assessed in turn.

Release Category A: Seismic Collapse of Containment

Q.19. Is there a case for regulatory action to reduce the seismic fragility of the containment at either Indian Point Unit?

A.19. The accident sequence with the most severe offsite consequences at either unit was identified in the IPPSS and the staff testimony to be one in which an earthquake causes failure of the containment. Although such accidents are assessed as being extremely unlikely, ways to reduce the vulnerability might prove to be inexpensive. The original edition of the IPPSS attributed the vulnerability of the Unit 2 containment to the slumping of the backfill into the side of the containment building. Removing or better anchoring the backfill might possibly be cost-effective. The staff is aware of no reason to doubt that the seismic qualification of the containments at the two units satisfies the regulations.

Q.20. What would be the monetary value of substantially reducing the seismic fragility of the containments at Indian Point Units 2 and 3?

A.20. The elimination of Release Category A events, as assessed in the staff testimony on Commission Question One gives rise to the following present worth monetary benefit estimates over the remaining life of the units for risk averted:

Unit 2:
$886,000 (safety goal)
$372,000 (low economic estimate)
$514,000 (medium economic estimate)
$1,120,000 (high economic estimate)

Unit 3:
$46,100 (safety goal)
$18,700 (low economic estimate)
$25,800 (medium economic estimate)
$56,000 (high economic estimate)

The late relocation model of emergency response was employed in these estimates, since the initiating event is an extremely severe earthquake, which could plausibly compromise emergency response.

Q.21. What other evidence bearing upon the seismic fragility of the containment is available?

A.21. The licensee submitted a letter to the Board, Staff and parties to the hearing during the presentation of testimony on Commission Question One. It treats a reanalysis of the fragility of the containment at the Indian Point Units 2 and 3. It purports to show that the containments are very much less fragile, i.e. vulnerable to earthquake-induced failure, than was suggested in the IPPSS. The staff has not completed its review, but preliminary indications suggest that the calculation have merit.

Q.22. What course of action does the staff recommend concerning the seismic fragility of the containment?

A.22. The staff will complete its review of the licensees' submittal. In light of the modest value of further risk reduction attributable to reducing the fragility of the containment, the staff does not recommend further pursuit of the matter unless the review fails to find merit in the new licensee calculations.

Release Category B: Interfacing Systems LOCA.

Q.23. Is there a case for regulatory action to further reduce the susceptibility of either Indian Point Unit to interfacing systems LOCA?

A.23. Release Category B is the second most severe class of releases identified in safety profile of the Indian Point Units. It is dominated by the rupture of the two normally closed valves on the suction line leading from the reactor coolant system outside containment to the Residual Heat Removal System. The staff has found no reason to believe that the design and operation of this system is out of compliance with the regulations. Prior testimony has indicated in Section A above that the Indian Point Units are less susceptible to this class of accident than are most PWR plants. Staff testimony[1/] suggests that roughly 7% of the early fatality

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1/ See Table IV B.3 in the staff testimony on Commission Question One.

risk posed by the two-reactor station originates in this class of accident. It has very much lower importance to other offsite radiological risks. Licensee testimony on Commission Question One suggests a more prominant role for interfacing systems LOCA in the early fatality risk, not because they find it to be more risky in absolute terms than does the staff, rather, it has higher importance in the licensee's testimony because their assessment suggests that other contributors to early fatality risk are smaller than the staff calculations suggest.

Several steps have already been taken to minimize the vulnerability of the units to interfacing systems LOCA. The Directors Order of February 11, 1980 called for additional surveillance on the check valves - another site for interfacing systems LOCA - of the ECCS system. Following the preparation of the IPPSS the licensees initiated a program of surveillance of the particularly important RHR suction valves, to be conducted during refueling outages. These considerations suggest to the staff that all reasonable and prudent actions to minimize the risk posed by interfacing systems LOCA have been taken. Nonetheless, it is wise to look at the value of additional actions.

Q.24. What value do the risk reduction benefit algorithms place upon the elimination of the residual risk of interfacing systems LOCA?

A.24. The value of entirely eliminating the residual risk posed by interfacing systems LOCA at the RHR suction line is assessed to be:

Unit 2: \$17,000/year or \$255,000, present worth (safety goal)
9,700/year or \$146,000, present worth (low economic algorithm)
11,600/year or \$174,000, present worth (medium economic algorithm)
18,700/year or \$280,000, present worth (high economic algorithm)

Unit 3: \$16,000/year or \$244,000, present worth (safety goal)

The economic algorithms give results roughly the same as those for Unit 2.

Q.25. What inferences does the staff draw from this evaluation?

A.25. It appears that design changes to further reduce the interfacing system LOCA contribution to risk are unlikely to be cost-effective. The addition of a third normally closed valve in series with the two whose rupture initiates the accident would not eliminate the risk but only reduce it, would introduce attendent risks (the valve might not always open where the RHR system is needed) and would likely cost more than the value of the risk reduction (if any) achieved.

6. Release Category C: Slow Overpressure Failure of Containment.

Q.26. Is there a case for regulatory action to further reduce the susceptibility of either Indian Point Unit to slow overpressure failure of the containment?

A.26. A large part of the offsite radiological risk originates in Release Category C events. Roughly 60% of the early fatalities and 90% or more of other off-site radiological consequence measures can be traced to accidents entailing loss of core cooling together with loss of containment cooling, i.e., damage state E. In 40% of the instances, damage state E leads to slow overpressure failure of containment, according to the staff assessment in the Commission Question 1.

A very substantial reduction in the risk has already been achieved by the four fixes described above. Then, too, there are a number of reasons to believe that the staff assessment of the residual risk is pessimistic. The most important of these are: 1) the staff has not credited the strengthening of the control room ceiling at either plant, which is important to the seismic contributers to Release Category C, 2) the staff acknowledges conservatisms in the after fix assessment of hurricane contributions, 3) full compliance with the fire protection rule has not been credited, and 4) if a reduction in source terms is warranted at all, it is particularly likely to be applicable to the late overpressure failure mode of containment, which has particularly long characteristic times to develop into a release.

The comparative risk assessment in Section A of the testimony above indicates that the contribution of "internally-initiated" accidents to Release Category C is well below average in individual risk and is probably roughly average in societal risk, compared with other plants licensed to operate by the Commission. Nonetheless, the ambiguities surrounding the comparative analysis of "externally-initiated" accidents, and the importance of Release Category C in the risk profile of the two plants suggest that it is worth entertaining the hypothesis that further risk reduction might prove to be cost-effective, although no firm commitment to order fixes should be made until the staff has had the opportunity to remove the unduely conservative elements in the value estimates.

Q.27. What fixes might be instituted to lower the risk contribution of Release Category C?

A.27. Both prevention options and mitigation options are likely to be feasible. Prevention might take the form of alterations narrowly targeted to lower the dominant contributors to Damage State E similar to those already implemented. Theoretically an add-on, dedicated decay heat removal system could be implemented, which could be designed to function as an alternate means of core cooling for a broad array of Damage State E accidents. However, such an add-on is believed to cost of the order of one hundred million dollars, and no design for a control and actuation system for such an add-on has yet been suggested that is free of attendent risks. Therefore, the add-on decay heat removal conception does not look promising. Mitigation conceptions along the lines suggested by Dr. James Meyer in the staff testimony on Commission Question 1, Part III B, also look to be feasible.

Q.28. What are the estimates of the value of the risk reduction afforded by the mitigation strategy described by Dr. Meyer?

A.28. The several valuation algorithms, applied to the release category frequency changes described in Dr. Meyer's Table III B 4 (page III B-31) for the realistic mitigation strategy are these:

Unit	Annual Value	Present Worth	Algorithm
2	$1,388,000	$20.8 million	Safety goal
"	$ 430,000	$ 6.5 million	Low economic algorithm
"	$ 528,000	$ 7.9 million	Medium economic algorithm
"	$ 838,000	$12.7 million	High economic algorithm
3	$ 772,000	$11.6 million	Safety goal
"	$ 220,000	$ 3.3 million	Low economic algorithm
"	$ 270,000	$ 4.0 million	Medium economic algorithm
"	$ 430,000	$ 6.3 million	High economic algorithm

Note that the economic estimates for valuing risk reduction yield lower estimates, applied to the mitigation strategy, than does the safety goal valuation formula. This pattern, which has not occurred in applications of the valuation guides to accident prevention, comes about because mitigation strategies do not affect the replacement power or substantially alter the costs of on-site cleanup.

Q.29. What regulatory inferences do you draw from the value estimates of the mitigation strategy?

A.29. It is clear from the estimates of the value of mitigation that the more expensive mitigation conceptions are not cost-effective at reducing the risk as assessed in the staff testimony on Commission Question One. A controlled, filtered vent (Contention 2.1a) or an add-on, fully qualified separate containment structure (Contention 2.1d) would cost substantially more than the risk reduction value would warrant. If Dr. Meyer's mitigation concepts could be implemented for as little as ten million dollars, the conception would fall in the gray area in which the uncertainties in the PRA estimates render it unclear whether the fix were cost-effect or not. However, it looks likely that Dr. Meyer's mitigation conception would cost several tens of millions of dollars, and the staff assessment of the value will decline when credit is given for all the installed or expected prevention fixes. Thus Dr. Meyer's conception does not look promising.

An alternate, diverse containment spray system might prove to be cost-effective. It is also plausible that a very economical mitigation conception might be found. For example, a "Tee" junction and appropriate valves might be added to one of the containment spray header pipes to enable a fire truck or other mobile pumping system to be rigged up to operate the containment sprays in some Damage State E events. Such a design would be less effective than Dr. Meyer's conception, and thus offer less value, but its costs might well be low enough to warrant the expenditure.

The staff proposes to give further consideration to such low-cost mitigation conceptions as well as further prevention fix options in its generic study of severe accident policy options.

Q.30. What value might accrue to further prevention fixes to lower the likelihood of Damage State E, and in so doing, lower the likelihood of Release Category C at the Indian Point Units?

A.30. A conservative bound on the value of further prevention fixes targeted on contributers to Damage State E can be obtained by applying the value algorithms to the elimination of Damage State E. Note that this exaggerates what further prevention fixes might really be worth in part because of the acknowledged conservatisms in the staff assessment of Damage State E and because practical prevention fixes could not be expected to fully prevent its occurrence. Perhaps 10% to 50% of the estimates below might realistically be within reach.

The value of eliminating Damage State E, as assessed by the staff for the "after fix" design and the evac reloc/late reloc models of emergency response are as follows:

Unit	Annual Value	Present Worth	Algorithm
2	\$1.54 million	\$23 million	Safety goal
"	\$1.12 million	\$17 million	Low economic estimates
"	\$1.23 million	\$19 million	Medium economic estimates
"	\$1.60 million	\$24 million	High economic estimates
3	\$820 thousand	\$12 million	Safety goal
"	\$550 thousand	\$ 8 million	Low economic estimates
"	\$610 thousand	\$ 9 million	Medium economic estimates
"	\$840 thousand	\$13 million	High economic estimates

It appears that expenditures up to a few million dollars may plausibly be cost effective if applied to better prevention of accident sequences leading to Damage State E. The studies to identify the fixes and translate them into detailed plans might well cost as much as their value.

7. Release Categories D through I: Other core melt accident risk contributers

Q.32. Is there a case for regulatory action to lower the frequency or mitigate the severity of accidents in Release Categories D through I?

A.32. One can obtain clues to the incentives to further reduce the offsite radiological risk posed by Release Categories D through I by tabulating the value, according to the safety goal benefit algorithm, of entirely eliminating each release category. For unit 2 these value estimates are:

Release Category	Emergency Response	Frequency (per year)	Annual Value (dollars)
D	evac reloc	0	$ 0
	late reloc*	1.01 (-6)	$ 22,700
E	evac reloc	0	$ 0
	late reloc	1.64 (-7)	$ 3,600
F	evac reloc	5.03 (-6)	$ 97,000
	late reloc	2.2 (-7)	$ 4,000
G	evac reloc	4.3 (-8)	$ 180.
	late reloc	3.02 (-7)	$ 1,400
H	evac reloc	2.15 (-5)	$ 11,000
	late reloc	6.27 (-5)	$ 36,000
I	evac reloc	2.58 (-6)	$ 18
	late reloc	2.21 (-4)	$ 1,700
Total RC D through I			$178,000/yr.
Present worth:			$2.7 million

*See next page for footnote.

For Unit 3 the calculation shows the values to be:

Release Category	Emergency Response	Frequency (per year)	Annual Value (dollars)
D	evac reloc	1.0 (-6)	$ 21,800
	late reloc*	0	$ 0
E	evac reloc	1.0 (-7)	$ 2,100
	late reloc	0	$ 0
F	evac reloc	6.1 (-6)	$114,000
	late reloc	9.6 (-8)	$ 2,000
G	evac reloc	3.3 (-7)	$ 1,400
	late reloc	1.3 (-8)	$ 65
H	evac reloc	6.4 (-5)	$ 35,000
	late reloc	6.2 (-6)	$ 3,900
I	evac reloc	2.5 (-4)	$ 1,900
	late reloc	1.7 (-6)	$ 14
Total RC D through I			$183,000/yr.
Present worth:			$2.7 million

*The late reloc emergency response model is employed for accidents triggered by regional non-nuclear disasters that could compromise emergency response.

For both units, the first and third ranked release categories are F and D respectively. The origin of both can be traced to containment failure due to hydrogen combustion. A mitigation strategy such as deliberate ignition or inerting the containment atmosphere could direct such accidents into the comparatively benign release categories H or I. The value of such a plant alteration, if nearly 100% effective, would be roughly $100,000 per unit year, or roughly $1.5 million over the life of each unit. Glow plugs installed in containment as deliberate, distributed ignition sources, might be installed for very roughly this amount. Thus the glow plug conception falls in the gray area of ambiguous cost-effectiveness. Other concepts for hydrogen control appear to be more expensive.

Since the safety goal valuation algorithm is known to give conservative estimates compared with the economic algorithms when applied to mitigation, it is illuminating to look at the values of deliberate ignition projected by the economic algorithms. This has been done for Release Category F at Unit 2. The elimination of the offsite radiological risk posed by RF is estimated to be $250,000 (Low), $330,000 (Medium), and $590,000 (High) over the life of the unit.

The cost effectiveness of hydrogen control looks less promising according to these algorithms. Dr. Meyer's research has also indicated that there may be significant attendent risks associated with the hydrogen control conceptions. Thus hydrogen control may warrant further generic consideration, but no concept emerges as clearly desirable at this time.

The remainder of the monetary incentive to reduce the risk posed by release categories D through I originates in the comparatively benign release category H. No offsite early fatalities and virtually no early injuries or offsite land interdiction is expected of Release Category H accidents. Very few latent cancers or genetic effects are projected. The offsite radiological risk posed by Release Category H is dwarfed by the expected on-site losses: replacement power and site cleanup.

Note that the principal reason that the overall core melt frequency is estimated by the staff to be over the safety goal design guideline of 1 in 10,000 per reactor year at both Indian Point Units is the substantial frequency of Release Category I events. These represent core melt accidents with no containment failure beyond elevated release rates. The central estimate of the benefit of eliminating such accidents from the risk profile of the plant, according to the safety goal algorithm of $1000 per person-rem averted, is less than $2000 per unit year, i.e. less than $30,000 in present worth. Clearly the staff analysis suggests that there is a negligible incentive, orginating in off-site radiological risk, to make these accident sequences less likely or less severe. If the staff has erred in its analysis that these accidents would be well-contained, there are two alternatives. Perhaps a higher percentage of these accidents might result in basemat melt through than Dr. Meyer's analysis indicates. Were this the case, some of these accidents would shift to Release Category H. There is very little off-site radiological risk attached to Release Category H, so we would still be left with the conclusion that public health and safety does not - by itself - warrant substantial expenditures to reduce these comparatively large contributors to the core melt frequency. The other way the staff analysis might be in error lies in the assessment of the ability of the containment to survive hydrogen burns. If a higher fraction of scenarios entailing core melt and hydrogen burns were found to breach containment than Dr. Meyer's analysis suggests, then some of the conparatively frequent Damage State I events would shift to Damage States F or possibly D. For these, as noted above, the incentive for lowering the risk is small but not negligible.

When we entertain the hypothesis that these accidents might not be so benign in off-site radiological consequences as our analysis suggests, we find that the sensitive assumption is the capability of the containment to survive hydrogen burns. Were we wrong, and I do not think we are likely to be, then we would find an amplified incentive to control hydrogen burns or to prevent such accident sequences. These phenomina will be further studied in generic reactor safety research.

Note that Release Categories H and I combined have an estimated frequency of roughly 3×10^{-4} per unit year at each Indian Point Unit. Replacement power costs and on-site cleanup would cost many billions of dollars were one of these accidents to happen. The economic algorithm suggests that the value of perfect prevention of these accidents would be roughly $2.3 million per unit year or $34 million, present woth at each unit. There may well be cost-effective ways the licensees could better protect their investment in the facility by making such accidents less likely, but public health and safety considerations do not appreciably add to these incentives.

Q.33. Does this conclude Appendix 2 and your testimony?

A.33. Yes.