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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

---:

IN THE MATTER OF:	: Docket Nos.
CONSOLIDATED EDISON COMPANY OF	: 50-247 SP
NEW YORK (Indian Point Unit 2)	:
POWER AUTHORITY OF THE STATE OF	: 50-286 SP
NEW YORK)Indian Point Unit 3)	:

---:

Westchester County Courthouse

111 Grove Street

White Plains, N.Y.

Tuesday, April 5, 1983

The hearing in the above-entitled mat convened, pursuant to notice, at 9 a.m.

BEFORE

TRO1

JAMES GLEASON, Chairman

Administrative Judge

ADD:
J. ARON H-1009
J. Whetstine EW-439

OSCAR H. PARIS

Administrative Judge

FREDERICK J. SHON

Administrative Judge

A P P E A R A N C E S:

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On Behalf of the Federal Emergency Management Agency

STEWART GLASS, ESQ.

On Behalf of the Intervenors

Council of the City of New York

CRAIG KAPLAN, ESQ.

Friends of the Earth, Inc., and New York City Audubon Society

RICHARD HARTZMAN, ESQ.

New York Public Interest Research Group

JOAN HOLT, ESQ.

AMANDA POTTERFIELD, ESQ.

JUDITH KESSLER, ESQ.

C O N T E N T S

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JUDGE GLEASON: Shall we proceed, please? Ms. Moore?

MS. MOORE: Your Honor, the staff calls to the stand Dr. James F. Meyer and Trevor Pratt.

JUDGE GLEASON: It's been some time. I have forgotten. Were you sworn?

MS. MOORE: Yes. Both witnesses were previously sworn.

Whereupon,

DR. JAMES F. MEYER

WILLIAM TREVOR PRATT

having previously been sworn by the Administrative Law Judge, testified as follows:

DIRECT EXAMINATION BY MS. MOORE:

Q. Dr. Meyer, would you please state your name and business address for the record?

A. (Witness Meyer) My name is James F. Meyer. My r business address is Nuclear Regulatory Commission, Washington, D.C..

Q. And would you please state your position with the NRC?

A. (Witness Meyer) I am a Senior Task Manasger responsible for the reactor system and containment system portions of severe accident analysis for resk assessment.

Q. Dr. Pratt, would you please state your name and business address for the record?

A. (Witness Pratt) William Trevor Pratt, Building 130, Brookhaven National Laboratory, Upton, New York.

Q. Would you please state your position?

A. (Witness Pratt) I am a group leader of the Accident Evaluation Group within the Department of Nuclear Energy

Q. Gentlemen, do you have before you a copy of the document entitled Direct Testimony of James F. Meyer and W. Trevor Pratt concerning Commission Question One?

A. (Witness Meyer) Yes, I do.

A. (Witness Pratt) Yes.

Q. Was this testimony prepared by you or did you participate in its preparation?

A. (Witness Meyer) Yes, it was prepared by us.

Q. Do you have any additions or corrections to this testimony?

A. (Witness Meyer) Yes, I do. In an errata sheet dated February 10, 1983, there are a number of corrections so indicated.

In addition, there is a typographical error on page 40 of our testimony, line 11 from the bottom of the page, the value 0.4 should read 0.5, and following that the parenthetical --

JUDGE GLEASON: Which page is that?

A. (Witness Meyer) Page 40, Roman Number 3.B-40.

JUDGE GLEASON: Roman Number 3.B-40? And where is that?

A. (Witness Meyer) The 11th line from the bottom. The value 0.4 should read 0.5 and parenthetical percentage should read 50 percent. That's the extent of my errata.

Q. With these changes to your testimony is it true and ccrrect to the best of your knowledge information and belief?

A. (Witness Meyer) Yes, it is.

A. (Witness Pratt) Yes, it is.

Q. Do you adopt it as your testimony in this proceeding?

A. (Witness Meyer) Yes, I do.

A. (Witness Pratt) Yes, I do.

MS. MOORE: Copies of this testimony have be delivered to the parties, Board and court reporter.

I move that it be received and bound in the transcript as though read. Dr. Meyer's professional qualifications were previously bound into this record.

JUDGE GLEASON: Is there objection?

Hearing none, the testimony of the witnesses will be received into evidence and bound into the record as if read.

(Bound testimony follows.)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket Nos. 50-247-SP
CONSOLIDATED EDISON COMPANY	)	50-286-SP
OF NEW YORK (Indian Point, Unit 2)	)	
	)	
POWER AUTHORITY OF THE STATE	)	
OF NEW YORK (Indian Point, Unit 3)	)	February 10, 1983

ERRATA SHEET
FOR
DIRECT TESTIMONY OF JAMES F. MEYER AND TREVOR PRATT
CONCERNING COMMISSION QUESTION 1

P.III.B-10-Line 7 DELETE "nine"

P.III.B-18-Column 4 CHANGE - "0.1" TO - "0.01"

P.III.B.-19-Line 6 CHANGE - "LFC" TO - "LF"

P.III.B-25-In the Column headed Release category B, the value in the line marked Release Time, CHANGE - "1" TO - "2" and the value in the line marked Release Duration CHANGE - "0.5" TO - "1.0"

P.III.B-38 and P.III.B-39 ADD as a footnote "values shown at the top of the bars are for fraction of cesium released"

P.III.B-45 DELETE the two lines immediately following Table III.B-8

P.III.B-58-Line 11 of Answer-25, CHANGE - "releases. When" TO - "releases, with the exception of the Ruthenium. When"

P.III.B-59-in the Column headed "Ru", the cross-hatched area should extend upward to a value of 0.21.

NOTE: Correction pages for pages III.B-32, III.B-33, III.B-43 and III.B-44 was submitted to the Board and Parties by, letter dated, February 2, 1983.

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 No. 2)	)	Docket Nos. 50-247 50-286
	)	
POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	)	

DIRECT TESTIMONY OF JAMES F. MEYER AND W. TREVOR PRATT
CONCERNING COMMISSION QUESTION 1

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

A.1 My name is James F. Meyer. U.S. Nuclear Regulatory Commission, Washington, D.C.

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

A.2 I am a Senior Task Manager responsible for the reactor system and containment system portions of severe (core-melt) accident analysis for risk assessment.

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

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

Q4. Please state your name and business address for the records Dr. Pratt.

A.4 My name is W. Trevor Pratt. Brookhaven National Laboratory (BNL), Upton, New York.

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

A.5 I am the Group leader for the Accident Evaluation Group in the Division of Engineering and Risk Assessment at BNL. As such I am responsible for the technical management of the USNRC Technical Assistance program at BNL in the area of severe accident analysis.

Q.6 Have you prepared a statement of your professional qualifications?

A.6 Yes, I have prepared a Statement of my professional qualifications attached to this testimony.

Q.7 What is the purpose of your testimony?

A.7 The purpose of this testimony is to address portions of Commission Question 1. The testimony presents the staff analysis and assessment of severe accident phenomena, containment building failure modes, and radiological releases from the containment buildings for Indian Point Units 2 and 3 that could result from coremelt accidents. In addition, analysis and assessment is presented of the changes in containment building failure modes and radiological releases that could result from potential design changes that mitigate the consequences of coremelt accidents. Much of the testimony in this section is based on information from "Preliminary Assessment of Core-Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating their Effects" (NUREG-0850, Vol. 1).

Q.8 Please outline the steps involved and the general approach taken by the staff in analyzing the containment buildings and mitigation features.

A.8 The purpose of this analysis and assessment is to determine the performance capabilities and failure characteristics of the Indian Point containment buildings under severe accident conditions; to generate data describing the radionuclide releases from the containment buildings for the environmental consequence evaluation based on the coremelt accident sequences determined in the testimony of Mr. B. Buchbinder, Mr. R. Budnitz and Mr. Sanford Israel; and to provide the bases for considering mitigation strategies. This analysis is accomplished in 7 steps:

Step 1. The plant damage states derived in the testimony of Mr. B. Buchbinder, Dr. R. Budnitz and Mr. Sanford Israel for internal and external events are combined into one set of plant damage states characterized by the time of core melt, the condition of the containment building, and the status of the containment building cooling capability.

Step 2. These plant damage states are analyzed using the MARCH computer code to determine the coremelt accident progression and the containment building loading and failure characteristics with and without mitigation strategies.

Step 3. Containment building event trees are established which provide for a convenient cataloging of the key events as the coremelt accident proceeds.

Step 4. Based on the MARCH computer code analysis, plus independent evaluations of such items as containment building failure pressure and steam explosions, split fractions are assigned to the various branches on these event trees. The output of the event trees is a set of conditional probabilities associated with various containment building failure modes for a given core-melt accident sequence. The failure modes considered can be thought of in two categories: those for which the containment building function is initially effective and those for which the containment function is either bypassed or significantly compromised. The first category is made up of the following failure modes (using the notation of the "Reactor Safety Study" (WASH-1400)):

α (alpha) Steam explosion induced failures. Steam explosions can potentially generate vessel-component missiles which could then penetrate the containment building.

γ (gamma) Hydrogen burn induced failures. Burning hydrogen gas can generate sufficient pressures inside the containment building to cause an overpressurization failure. (Hydrogen burns can also cause the containment building to fail indirectly by causing the failure of engineered safety features needed to protect the building containment function.)

δ (delta) Failures induced by overpressurization of the containment building produced from generation of steam and noncondensible gases. The release of primary system energy in the form of steam, combined with the decay heat energy which produces more steam and noncondensible gases, can overpressurize the containment building, thus leading to failure.

ε (epsilon) Basemat penetration. Core materials interacting with the reactor cavity basemat can penetrate the containment building floor (basemat), thus releasing core materials and water into the environment.

The second category is made up of the following failure modes (using the notation of WASH-1400)

β (beta) Failure to isolate containment building. The coremelt accident occurs with containment building penetrations left open, thus considerably reducing the effectiveness of the containment building function. The conditional probability that this containment building isolation occurs is 10^{-3}, that is, one time in a thousand the containment building will not be isolated during a core-melt accident.

V The accident progression bypasses the containment building function completely. An example of this failure mode is the interfacing systems loss-of-coolant accident. It is due to the failure of barriers, such as check valves, that separate high pressure from low pressure systems. Direct access to the environment is obtained through the residual heat removal system piping.

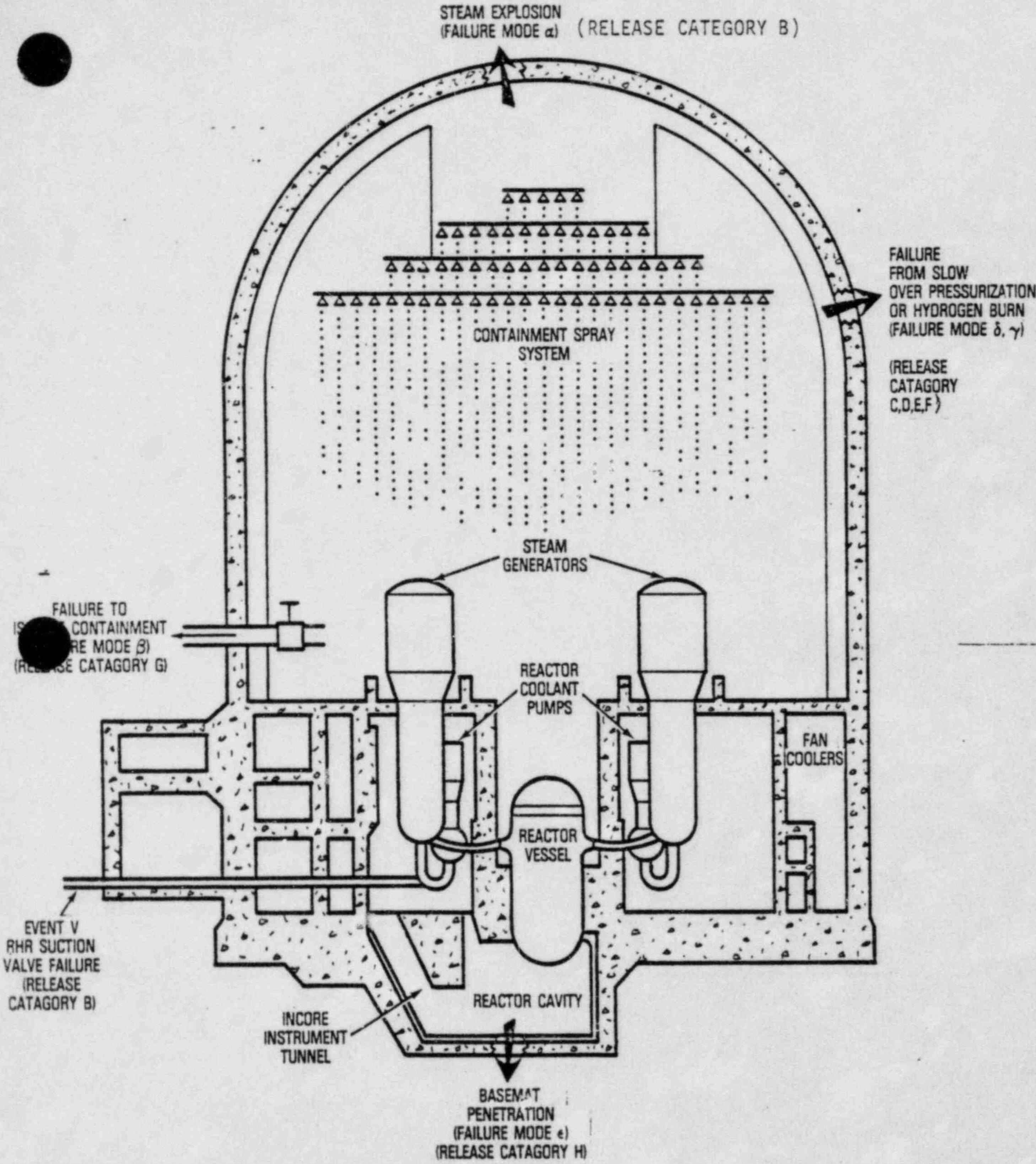
β* The initiation of a coremelt accident and concurrent failure of the containment building are caused by an external event, such as an earthquake. This failure mode (not considered in WASH-1400) is similar to a major "β" failure mode because it assumes a very large opening in the containment building.

TR This accident progression bypasses the containment by means of the steam generators. A core melt accident progression develops which is characterized by multiple steam generator tube ruptures (TRs) and failed (stuck open) secondary system pressure relief valves.

All these modes are considered in the staff analysis; however, only the first category of failure modes is treated in the containment event trees.* Failure mode categories are subcategorized according to the different times of containment failure and different conditions within the containment. Figure III.B.1 pictorially shows these various containment building failure modes (with the exception of β*, the seismically induced containment building failure and TR, the steam generator tube rupture failure mode). In addition, this figure also relates the failure modes to the release categories (a matter to be discussed later in this testimony).

Step 5. For each containment failure mode, a CORRAL computer code analysis yields the radiological release values at the point of containment building failure (or the radiological release at a leak rate of 1% of the containment building volume per day for the "no-failure" cases). These radiological release values are grouped into 9 release categories which cover the full spectrum of releases from severe

Since, for the second category of failure modes (β, V, β, TR) the containment function is already defeated, no containment event tree analysis is needed.



REACTOR BUILDING

Figure III.B-1 Indian Point Reactor Building Showing Potential Failure Modes

accidents that cause early containment building failures to accidents that cause minor leakage but that do not otherwise compromise the building's containment function.

Step 6. Similar containment building loading analyses are conducted with various mitigation features in place. The reduction or elimination of various containment failure modes due to the presence of mitigation features is reflected in changes in the probability for each of the 9 release categories. These changes are determined by using the containment event trees.*

Step 7. The release category output is then used to determine the actual risk reduction afforded by the mitigation features by performing CRAC computer code consequence analyses, as described in the testimony of Dr. S. Acharya.

Q.9 Please continue with a detailed discussion of the containment building loading and failure evaluation following the outline given above. Start by discussing the grouping of plant damage states derived from the testimony of Mr. B. Buchbinder, Mr. R. Budnitz and Mr. Sanford Israel.

A.9 Step 1. The grouping of plant damage states from internal, fire and external events is shown in Table III.B.1. In this table the damage states given in Mr. Israel's testimony (internal events) are added to the damage states given in Mr. Buchbinder's testimony (fires) and Dr. Budnitz's testimony (external events) for each unit.** The external-event damage states, the "fire" damage states, and the internal-event damage states can be combined in this fashion because the subsequent accident progression within the containment is relatively insensitive to the accident initiators once the status of the containment has been determined. Note that the

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*It should be noted that in general the second category of failure modes, for which there are no event trees, are unaffected by the presence of mitigation features described in this testimony.

**The memorandum which describes the assembling process for the damage states listed in Table III.B.1 is memorandum from F. Rowsome to J. Meyer, "Damage State Liklihood For Indian Point," dated December 2, 1982.

Table III.B.1 Indian Point damage state frequencies

Damage State	Unit 2						Unit 3					
	Before Fix*			After Fix*			Before Fix**			After Fix**		
	INT***	LOSP†	RD††	INT	LOSP	RD	INT	LOSP	RD	INT	LOSP	RD
Z	0	0	7(-7)	0	0	7(-7)	0	0	3.5(-8)	0	0	3.5(-8)
V	4(-7)	0	0	4(-7)	0	0	4(-7)	0	0	4(-7)	0	0
E	4(-4)	1.6(-5)	3.2(-4)	2.4(-5)	1.6(-5)	4.3(-5)	3.6(-4)	1.5(-6)	1.2(-5)	2.5(-5)	1.5(-6)	1.2(-5)
EC	1.1(-5)	0	0	1.1(-5)	0	0	neg.	0	0	neg.	0	0
EF	6.4(-7)	0	0	6.4(-7)	0	0	neg.	0	0	neg.	0	0
EFC	1.3(-4)	2(-5)	6(-9)	1.3(-4)	2(-5)	6(-9)	2.0(-4)	3(-6)	1.2(-6)	2.0(-4)	3(-6)	1.2(-6)
LF	1(-4)	0	0	1(-4)	0	0	1(-4)	0	0	1(-4)	0	0
SGTR	2(-6)†††	0	0	2(-6)	0	0	2(-6)	0	0	2(-6)	0	0
TOTAL	6.4(-4)	3.6(-5)	3.2(-4)	2.7(-4)	3.6(-5)	4.3(-5)	6.6(-4)	4.5(-6)	1.3(-5)	3.3(-4)	4.5(-6)	1.3(-5)
GRAND TOTAL		1(-3)			3.5(-4)			6.8(-4)			3.5(-4)	

*Fixes for Unit 2 include a) reduced seismic fragility, b) reduced fire vulnerability, and c) anticipatory shutdown for hurricanes

**Fixes for Unit 3 are limited to reduced fire vulnerability

***INT = Internal events excluding those characterized by loss of offsite power

†LOSP = Events limited to those characterized by loss of offsite power (LOSP)

††RD = External events characterized as regional disasters (RS) (seismic and hurricane)

†††After the analysis in this testimony was completed, the value for core melt was changed upward to 4(-6) (see staff testimony on Board Question 2.2.1)

Table III.B.1 Indian Point damage state frequencies (continued): Definition of "damage state" designations

Plant Damage States	Designation
Containment Failure Prior to Core Melt	Z
Containment Bypass Via Interfacing Systems LOCA	V
Early Core Melt With No Containment Cooling	E
Early Core Melt: Sprays and Coolers Operational	EFC
Early Core Melt: Only Coolers Operational	EF
Late Core Melt: (Failure of ECCS in recirculation mode) with coolers operational	LF
Early Core Melt: Only Sprays Operational	EC
Containment Bypass Via Steam Generator Tube Rupture	SGTR

damage state probabilities are given both before and after the "external event fixes" described in previous testimony. Also note that the damage states are further separated into groupings which characterize the site evacuation capability, namely normal site evacuation capability and abnormal site evacuation capability due to "regional disasters." (This matter will be discussed in some detail by S. Acharya in testimony to follow mine.)

Q.10 Please summarize the results of the MARCH code analysis and independent analysis for the damage states mentioned above.

A.10 Step 2. Five of these eight representative damage states were analyzed in NUREG-0850.* A dominant damage state, "E," for both Units 2 and 3, is characterized by a small break LOCA coupled with failure of the emergency core cooling system (ECCS) injection and all containment heat removal systems (CHRS) systems. A typical small break LOCA results from failure of the reactor coolant pump seals.** The leakage rate through the failed pump seals was assumed in NUREG-0850 to be approximately 200 gal/min, compared with the rate of 1200 gal/min assumed in the IPPSS. The faster leakage rate assumed in the IPPSS would shorten the time to uncover and degrade the core and to cause vessel failure relative to the analysis described in NUREG-0850. However, because the characteristics of the accident sequence will be similar, the analysis presented in NUREG-0850 will form the basis of our discussion. For accident sequences in which water from the refueling water storage tank (RWST) would not be injected into the containment, and with failure of all CHRS, the cavity would not be flooded with copious amounts of water at vessel failure and, further, reflux of water into the reactor cavity could not be assumed. This implies that the analysis presented in Section 3.2.2.2(1) of NUREG-0850 which assesses the dry-cavity cases is applicable to damage states designated "E."

*The remaining three damage states, designated as Z, V, and SGTR here, were assessed subsequent to the publication of NUREG-0850.

**This accident is equivalent to the TMLB'S sequence discussed in Sections 3.2.2.2(1) and 3.2.3.2(1) of NUREG-0850.

Figure III.B.2 shows a typical pressure history for an "E" damage state with a small break LOCA coupled with failure of ECCS injection and all CHRS systems. Initially following reactor scram, the containment building remains at operating pressure because the primary system energy is released through the steam generators. If there is no feedwater available to the steam generators they will eventually boil dry. Also during the total-loss-of-ac-power accident, the reactor coolant pump seals cannot be cooled. Under these circumstances, a small LOCA induced by failure of the reactor coolant pump seals could occur. In NUREG-0850, pump seal failure was assumed to occur one hour after scram. The pressure history in Figure III.B.2 reflects this assumption. After the steam generators boil dry or the pump seals fail, release of the primary system water inventory will start. If the steam generators boil dry first, the power operated relief valve (PORV) will lift and relieve primary system water at the set point of the valves. As soon as the pump seals fail, the primary system will start to depressurize and primary system water will be released through the failed pump seals.

When primary system water is released to the containment building, it flashes to steam. Since no active containment heat removal is assumed for this accident sequence, the steam partial pressure causes the pressure in containment to rise. Eventually, without ECCS injection, the core will uncover. The point at which the core is uncovered depends on the time at which the pump seals fail and on the leakage rate. In Figure II.B.2 we indicate the point at which the core is uncovered for a typical "E" damage-state accident progression. In NUREG-0850 sensitivity studies were provided on leakage characteristics.

When the core is uncovered it heats up and will eventually melt. We predict that it will take approximately one hour for the core to melt and slump into the bottom of the reactor vessel. When the molten core contacts water in the bottom vessel head, steam will be produced and released to the containment atmosphere via the failed pump seals. In the bottom of the reactor vessel the core materials will thermally attack the lower vessel head. With a relatively low primary system pressure, we predict that

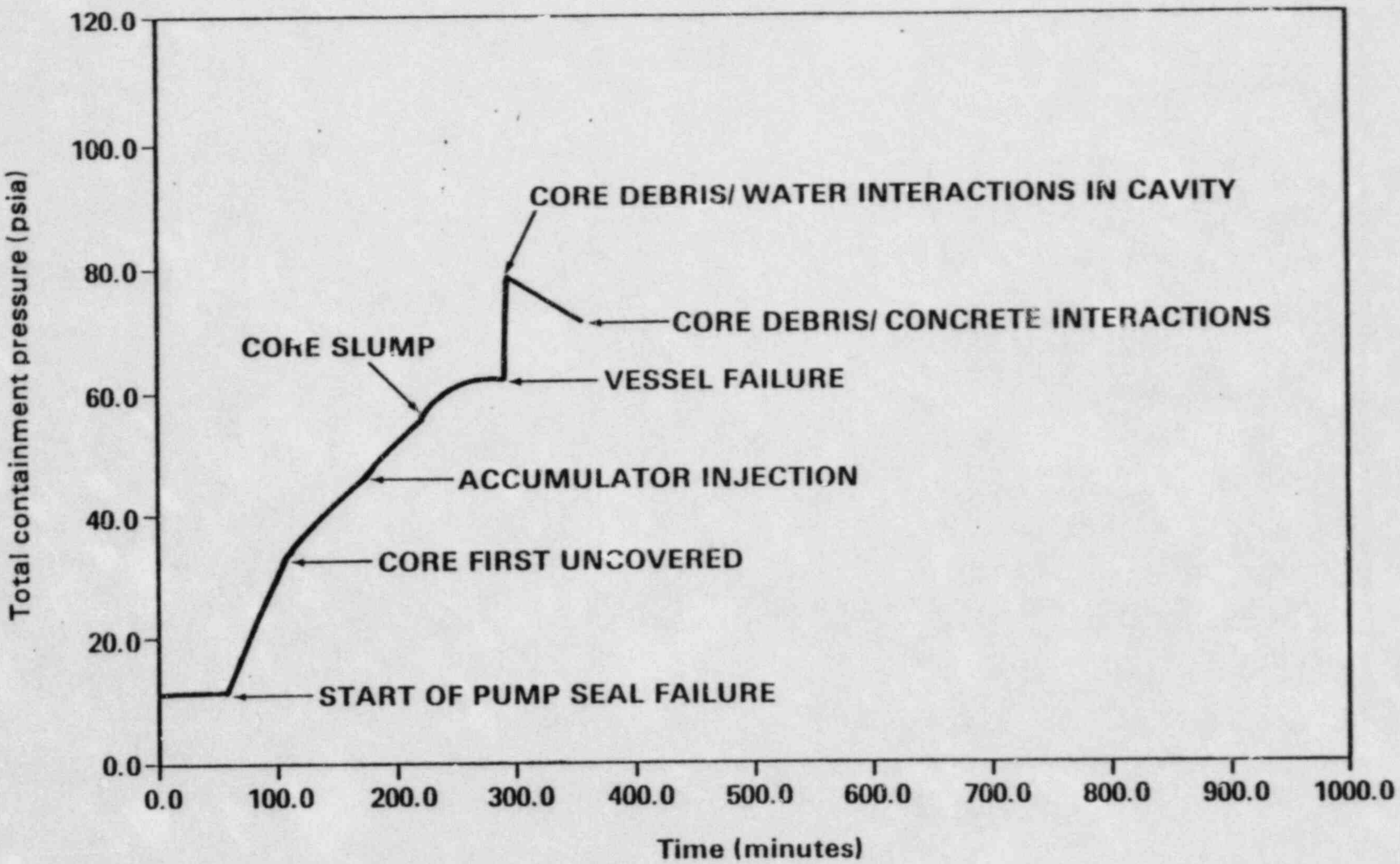


Figure III.B-2 Typical "E" Damage State Pressure History it will take approximately one hour for the core materials to degrade the bottom head to the point at which failure would occur.

When the reactor vessel fails, any residual primary system pressure is released to containment while the core materials are released into the reactor cavity. For this accident sequence, relatively low primary system pressures are predicted so that minimal blowdown forces result from vessel failure. Also, we noted above that a minimal amount of water would be expected in the reactor cavity at vessel failure. Water could reach the cavity after vessel failure from the accumulator tanks, provided high pressures prevent accumulator injection into the primary system prior to vessel failure. Accumulator injection occurs when the primary system pressure falls below 665 pounds per square inch absolute (psia). Primary system depressurization depends on the leakage rate, which in turn depends on the size and location of the break. Figure III.B.2 shows that for the assumed leakage rate, high primary system pressure does prevent injection of all accumulator water prior to vessel failure here. Some of the accumulator water discharges into the cavity after vessel failure. However, there is not sufficient water available in the cavity to bring the core debris into thermal equilibrium.

With limited water in the reactor cavity, extensive interactions will eventually occur between the core material and the concrete. As the core debris decomposes the concrete, water vapor and carbon dioxide are released. The water and carbon dioxide can oxidize metals in the core debris and form combustible hydrogen and carbon monoxide. However, for this particular accident sequence, the high partial pressures of steam and noncombustible gases render the containment atmosphere inert. Consequently, failure of containment due to burning of combustible gases is not a potential failure mode for this damage state.

Potential failure modes of concern are failure of the containment building by overpressurization caused by the release of steam and noncondensible gases from the concrete, or failure by basemat penetration. In the IPPSS it was assumed that failure would occur by overpressurization rather than by basemat penetration. However, in NUREG-0850 it was found that the type of concrete installed at the reactor site strongly influenced the potential for the two failure modes (Section 3.2.2.3-3.2.2.6). A limestone-type concrete, which releases large quantities of noncondensible gases, was calculated to fail the containment building by overpressurization prior to basemat penetration. However, for basalt-type concrete, which releases smaller quantities of noncondensible gases, it was not clear that overpressurization would occur prior to basemat penetration. Analyses of concrete specimens from Indian Point indicate a basalt-type concrete was used; hence we are not able to predict with certainty the timing and mode of containment building failure. The uncertainty of these two failure modes during core/concrete interactions is reflected in the branch point split fractions used in our containment event trees (see step 4 below).

Another class of damage states is characterized by the availability of ac power and, thus, the availability of containment building heat removal (either by sprays or fan coolers). For these damage states (EC, EF, EFC, and LF), the primary threat to the containment building comes from burning of combustible gases (principally hydrogen), either directly due to pressure transients, or indirectly by causing damage to ESFs. The hydrogen burning problem would be severe for either a dry or flooded reactor cavity, although there is an expectation of a smaller contribution from hydrogen combustion for a flooded cavity if a coolable debris bed is established shortly after vessel failure (about 1/2 hour). Basemat penetration is calculated for those sequences when a dry cavity is assumed. Although water is certain to slow down penetration times, it will not arrest the penetration unless a coolable debris bed is established for the core melt materials. The probability and timing of containment building failures by hydrogen burning are specifically treated in this testimony by the relevant branch points in the containment event trees.

The thermal loadings on the containment building are considered to be of secondary importance under saturated conditions within the containment building. Calculated thermal loadings for accident sequences that did not involve combustion or dry-cavity core/concrete interactions (350°F) are considered to be small. For sequences involving combustion, temperature loadings may be important in assessing equipment survivability and the integrity of containment building penetrations. Also for "dry-cavity" accident sequences where considerable noncondensible gases are produced, the super-heated environment which evolves may be important. These matters are discussed under the subject of uncertainties later in this testimony.

Steam explosions and any missiles they might generate that could penetrate the containment building (e.g., control rods and vessel head) were important considerations in WASH-1400 risk analysis. Potentially these missiles could cause a containment building failure for any of the damage states under consideration. The consequences from this failure mode ("α" in WASH-1400 notation) can be severe. In NUREG-0850, the probability of an "α-mode" failure is over 100 times smaller than was estimated in WASH-1400.* Our estimate is based on such considerations as the fraction of the core melt involved in the heat transfer process; the efficiency of the heat transfer process; the effect of the steam explosion on the vessel head and control rods; the ameliorating effects of the containment missile shield, and the resiliency of the containment building. The essential argument for the lower values is that reactor vessel geometry is not conducive to premixing large quantities of core materials--a prerequisite for large steam explosions. The probability for a given energy release is written as the product of (a) the probability of obtaining a premixture quantity of drops of molten core material in water commensurate with such energy yield and (b) the conditional probability that such a premixture will be triggered into a coherent explosion. These probabilities were evaluated for all the stages of the meltdown sequence. The upper bounds of a few hundred megajoules and a few thousand megajoules are projected for the in-vessel and out-of-vessel steam explosions, respectively. Even though

*An analysis by the NRC Office of Nuclear Regulatory Research, done independently of the analysis given in NUREG-0850, draws a similar conclusion, namely, that the best-estimate probability for the "α-mode" failure is 100 times smaller than estimates in WASH-1400. Note: M. L. Corradini and D. V. Swenson, Sandia National Laboratories, "Probability of Containment Failure Due to Steam Explosions Following a Postulated Core Meltdown in an LWR," U.S. NRC report NUREG/CR-2214, June 1981.

the probabilities are low, steam explosions are formally considered in the overall risk analysis and in the containment event trees described below.

Q.11 Please discuss the establishment of containment event trees.

A.11 Step 3. Containment event trees are convenient ways to organize and represent the progression of coremelt accidents from accident initiation on the left of the tree to the point of containment failure on the right of the tree. Containment event trees have been constructed for four representative damage states described in Table III.B.1: E, EFC,* EF, and LF. The trees use a ten-branch logic network to describe the progression of the various states. The questions asked at each of the ten branches are described in Table III.B.2. A yes-no answer is required for each of the ten questions. Each branch is assigned a split fraction which assigns a probability of a "yes" answer to the branch point question (the probability of a "no" answer is one minus the probability of a "yes" answer). The selection of the split fraction depends on the type of analysis presented in step 2 above. The convention used in the trees is that a positive response (yes) results in a path which moves to the top of the tree and a negative response (no) results in a path which moves to the bottom of the tree.

A positive response to any one of six questions (2, 3, 5, 8, 9 and 10) results in failure of the containment building by a variety of failure modes. Each of these failure modes results in a particular radiological release category (see step 5 below). For those paths which do not have a positive response for any of the six questions, the path will end in "no containment failure." However, it is important to note that "no failure" paths also result in releases of fission products to the environment because of containment leakage. Finally, for each individual tree, the conditional probabilities associated with the end points of the various paths through the tree (i.e., the right-hand column) should sum to unity. These conditional probabilities are then multiplied by the probabilities

*Because the EC damage state has a similar containment building loading history to EFC it is subsumed into the EFC damage state.

Table III.B.2 Containment event tree branch questions

Branch Division	Questions
1	Is there a substantial hydrogen burn prior to vessel failure?
2	Does containment fail prior to vessel failure ("γ" failure mode)?
3	Does containment fail by steam explosion generated missiles ("α" failure mode)?
4	Is the cavity flooded at vessel failure?
5	Does containment fail at the time of vessel failure ("δ" failure mode)?
6	Are CHRS* restored after vessel failure (restoration of ac power)?
7	Are containment building sprays operating?
8	Does containment fail by combustible gas burning ("γ" failure mode)?
9	Does containment fail by overpressurization ("δ" failure mode)?
10	Does containment fail by basemat penetration ("ε" failure mode)?

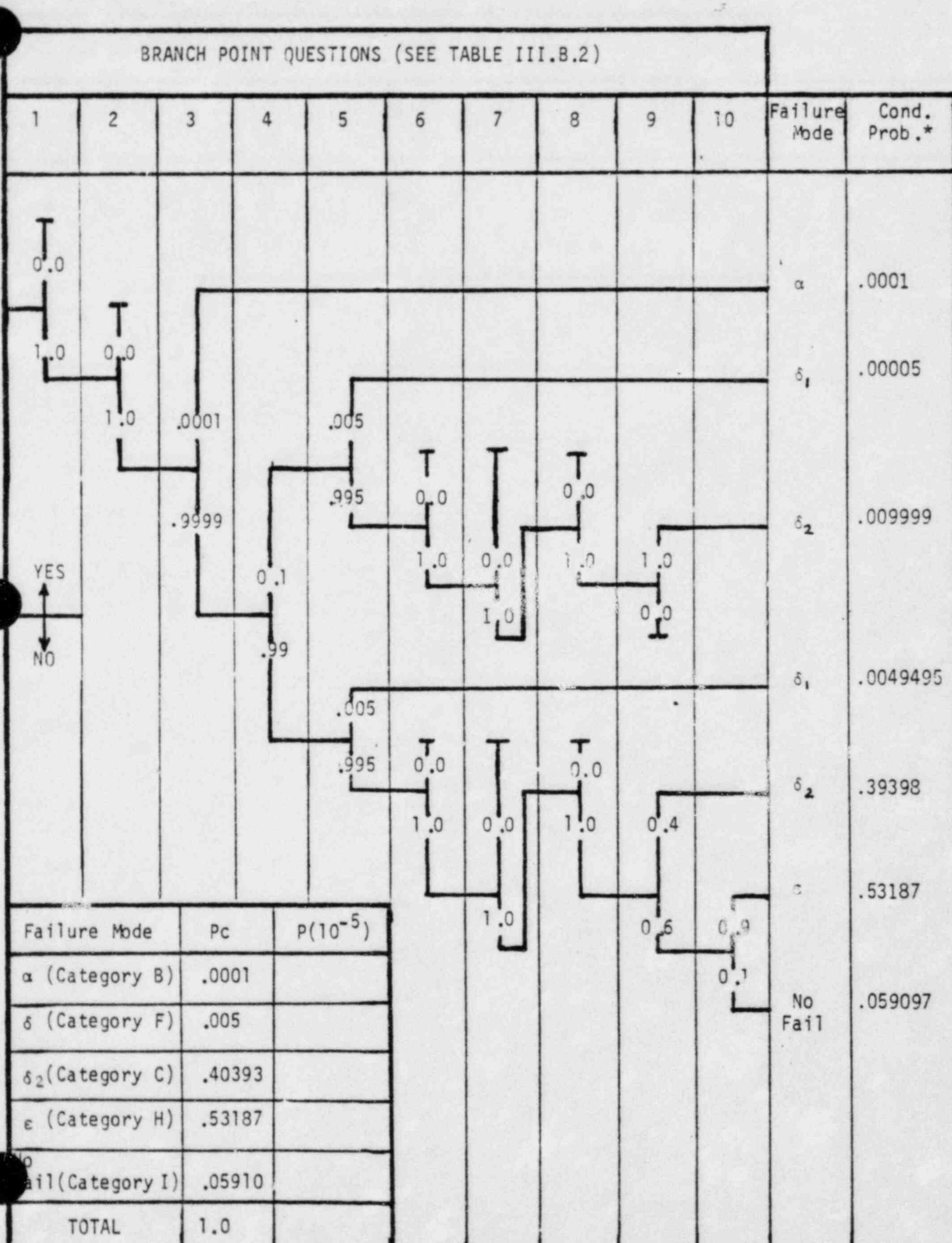
*CHRS: Containment Heat Removal Systems

for the accident sequences themselves as listed in Table III.B.1 to determine the probability of failure by the particular failure modes examined. A sample containment event tree is given in Figure III.B-3.

Q.12 Please continue your discussion of the event tree method by describing the analysis associated with each branch point.

A.12 Step 4. This overall explanation will be clearer if we examine a typical containment event tree, branch by branch, and describe the analytical process at each step. Note Figure III.B-3.

"E" DAMAGE STATE
FIGURE III.B-3 CONTAINMENT EVENT TREE



Failure Mode	Pc	$P(10^{-5})$
α (Category B)	.0001	
δ (Category F)	.005	
δ_2(Category C)	.40393	
ε (Category H)	.53187	
No Fail(Category I)	.05910	
TOTAL	1.0	

*The numberical values indicated here are only known to one or perhaps two significant digits. All additional digits are only included so as to audit that the conditional probabilities add up to unity.

The <u>first</u> point at which the branch could divide considers the question of large amounts of hydrogen generated and released to the containment building prior to vessel failure. Due to characteristics of the accident sequences analyzed in the MARCH computer code and independent calculations, it was determined that a substantial hydrogen burn would only take place for the "LFC" damage state prior to vessel failure. A substantial burn does not necessarily mean containment failure. In fact, a substantial early burn could be beneficial because it could prevent a more extensive burn later when more hydrogen and other combustible gases have been released to the containment building.

The <u>second</u> point at which the branch could divide considers containment building failure before vessel failure. For our analysis, the only important containment building failure mode at this branch is failure by hydrogen burn and then only for the "LF" damage state. (However, it should be noted that the B* containment failure mode, not considered in the event trees, also fails the containment prior to vessel failure). For any failure determination, the pressure at which the containment building fails must be calculated. The NRC analysis* determined that extensive yielding of structural members (reinforcing bars) for the Indian Point containment building would take place at 126 pounds per square inch gauge (psig) (141 psia). We have defined this as the staff failure pressure. For overpressurization from either hydrogen burns or from steam and noncondensible gas overpressurization, which generate pressures greater than 126 psig, the conditional probability for failure approaches 1.0. NRC analysis also determined that there would be extensive strain in the liner of the containment building at the intersection of the cylinder wall and foundation basemat at 118 psig, which could result in local liner failure and resultant leakage to the environment. For those events such as hydrogen burns, which are large but have pressure peaks of less than the 126 psig failure pressure, the following probabilities for containment building failure were used, based on considerations of the uncertainty of the

---

*Letter from F. Schauer, Chief, Structural Engineering Branch, DE, NRR to G. Mazetis, Acting Chief, Reactor Systems Branch, DIS, NRR dated January 11, 1982.

126 psig value and of leakage due to liner failure at the lower pressure. A 5 psi standard deviation value (noted as σ) was assumed, yielding the following probabilities for containment failure:

Probability of failure at:

126 psig = 0.98 (+2σ)
121 psig = 0.84 (+1σ)
116 psig = 0.50 (mean value)
111 psig = 0.16 (-1σ)
106 psig = 0.02 (-2σ)

For this branch (#2), the highest split fraction containment building failure probability determined was 0.01. The reason for this value is given in Appendix A of this testimony, "Discussion of Event Tree Branch Divisions for Hydrogen Burns."

The third branch division addresses the question of whether the containment building will fail by steam-explosion-generated missiles. As discussed above, we view this failure mode as having a low probability. The split fraction used in this study for all five sequences is 10^{-4} (0.0001) for a positive response at this branch.

The fourth branch division addresses the question of whether the cavity will be flooded at or after vessel failure and whether or not it will remain flooded thereafter. This question is important, as it will affect steam overpressurization, basemat penetration, aerosol generation, and hydrogen generation. The split fractions used for this branch are:

E flooded = 0.01
EF flooded = 0.01
LF flooded= 1.00
EFC flooded= 0.90

These split fraction probabilities are a function of the capability to inject the refueling water storage tank water into the containment building and on whether or not water is present in the cavity at the time of vessel failure. (Water will definitely be present at vessel failure for the LF damage state but perhaps not for the EFC.)

Branch divisions 5, 8, and 9 address containment building overpressurization failures (from burns or steam) relatively early in the accident (5) and relatively late in the accident (8 and 9). The hydrogen burn failures are of concern for the EFC, EF, and LF damage states. The steam overpressurization failures (actually a combination of steam and noncondensible gases) are of concern for the E state. The split fractions used are listed in the event trees.

The split fraction probabilities for containment building failure due to hydrogen burns range from 0.0001 to 0.15. These split fractions are derived from data about hydrogen production, its release to and mixing with the containment atmosphere, and its burning using the MARCH computer code and independent analyses. The background for this determination is given in NUREG-0850 while the specific procedure for determining the split fractions is given in Appendix A to this testimony.

The split fractions for containment building failure due to overpressurization from steam and noncondensible gases are a function of whether or not the reactor cavity is flooded. If the reactor cavity is flooded, containment failure by overpressurization is virtually assured and therefore a split fraction for containment failure is 1.0. But if dry, then the analyses in NUREG-0850 indicated that the pressures alone may not be sufficient to fail the containment building prior to basemat penetration; the branching probability split fraction assigned for this case is 0.4. Because, for this case, there exists considerable uncertainty, a sensitivity analysis is performed later in this testimony which considers the impact of this uncertainty on risk.

Branch division 6 addresses the question of restoration of ac power after core meltdown and reactor vessel failure. In particular, we are concerned with restoration of containment heat removal systems (CHRS). This branch is only relevant to the E damage state. At branch 6 a negative response of "1" is assigned to the split fraction for damage state E. The reason that there is no probability assigned here for power restoration is that the staff has determined that: either power will be restored prior to core-uncovery thereby preventing core degradation or, if not, then the probability of power restoration occurring early enough to affect the outcome of the "E" damage states is small. Restoration of ac power and thereby cooling virtually eliminates the possibility of an overpressurization failure but enhances the probability of hydrogen burn failure, as discussed below.

Branch division 7 addresses whether sprays are operating or activated. It is important to know if sprays are operating because they reduce the airborne radiological source term and have an ameliorating effect on hydrogen burns should they occur. (Either fan coolers or sprays can make the containment atmosphere combustible that had previously been rendered inert by steam and thus cause hydrogen ignition; this matter is addressed in branch 8). The following split fractions were used for branch division 7:

Split fractions for damage state:

E = 0.0
EF = 0.0
EFC = 1.0

The final branch division 10,* addresses the question of basemat penetration. In NUREG-0850, the analysis indicated that no basemat penetration would occur if a coolable debris bed could be established in the reactor cavity. If a coolable debris bed could not be established, basemat penetration is predicted to take about three days. For the flooded cavity

*Note that branch divisions 8 and 9 are discussed under branch division 5 above.

cases, a conditional probability for failure of 0.1 was assigned, reflecting the uncertainty that a coolable debris bed could, in fact, be established even with a flooded cavity. (This issue is assessed in Section 3.2.3.3 of NUREG-0850). For a dry cavity, a conditional probability for failure of 0.9 was assigned, reflecting the assessment that basemat penetration is the dominant course for the accident progression when no cooling is available.

In summary, four damage states are analyzed to determine the characteristic containment failure modes. Containment event trees (one for each of the four damage states) are used to catalog the accident progressions and, specifically, to determine conditional probabilities for the various containment failure modes. (These conditional probabilities are found in the right-hand column of the event trees.)

Q.13 Now that you have described the physical mechanisms which can cause containment building failure, please describe how you determine the radionuclide release categories for each of the failure modes.

A.13 <u>Step 5</u>. With all the containment building failure modes and related characteristics compiled, the next step in the procedure is to combine them into categories which describe the amount and composition of the radiological release at the time of containment building failure. This process is accomplished with the aid of the CORRAL computer code, which determines the radiological release fractions to the environment based on physical processes occurring in the containment building.

Output from the MARCH code related to core degradation, conditions in the containment building atmosphere, and leakage from the containment building are used as input to the CORRAL code. The CORRAL analysis follows the radiological release from the time the core is uncovered to the containment building blowdown at building failure. CORRAL calculates the integrated release fractions of eight radionuclide groups as a function of time. The CORRAL output defines the release for a given containment failure mode for a specific accident sequence. CORRAL calculations are performed for all possible containment failure modes for each of the damage states. This leads to a large number of potential release categories. In order to make further calculations managable it was determined that the nine release categories listed would adequately represent the large number of calculated releases.

The release categories shown in Table III.B.3 range from the most severe situations (Category A) to the release resulting from a 1% per day leakage (Category I) for the no-failure case. Each of the containment building failure modes for the damage states is assigned a release category, as pictorially indicated in Figure III.B-1 and listed in Table III.B-3. Table III.B.3 gives the data for all nine categories. The probabilities shown are determined by multiplying the probability for the damage state in question (note Table III.B.1) by the total conditional probability for the particular failure mode (note Figure III.B.3). The information in Table III.B.3 is exactly the data needed to perform the risk analysis (CRAC-code analysis) described in the testimony of Dr. S Acharya.

Q.14 What is the staff position regarding use of the CORRAL code?

A.14 The procedures and assumptions described here for determining the radionuclide release values (and described further in Section 3.4 of NUREG 0850) are essentially those used in WASH-1400 and in other PRA studies (e.g., Zion Probabilistic Safety Study and the IPPSS). More recent calculations with more mechanistic aerosol behavior codes have demonstrated that the CORRAL code, with standard input assumptions, do not fully consider the amount of aerosol agglomeration in the containment building atmosphere.* In addition, CORRAL does not address retention of aerosols in the primary system. The present data base, however, does not permit a quantification of the impact of the above considerations. We find little basis for the attempt to quantify these effects with probability distributions assigned to the various source terms in the licensee's IPPSS. Probability

*Technical bases for estimating fission product behavior during LWR accidents USNRC-NUREG-0772, June, 1981.

Table III.B.3 Radiological Releases f[illegible]he Containment Building - CRAC Input

Release Category	A	B	C	D	E	F	G	H†	I†
Associated Failure Mode	β*	V, α	δ, TR	γ	γ	γ	β	ε	NF**
Release Time (hours)	3	1	13	9.4	12	3.0	2	72.0	2††
Release Duration (hours)	2.0	0.5	0.5	0.5	0.5	0.5	8.0	8.0	8.0
Release Energy (BTU/hr x 10^6)	5.0	0.5	98	137	180	180	0.3	0	0
Warning Time (hours)	1	1	8	1	1	1	1	67	1
Release Fraction (fractions of total core inventory)									
Xe-Kr	1.0(0)##	1.0(0)	9.6(-1)	1.3(-1)	8.5(-1)	1.4(-1)	1.0(0)	7.0(-1)	5.0(-4)
I-Br	8.0(-1)	7.0(-1)	9.8(-2)	1.0(-1)	1.0(-1)	7.8(-2)	2.0(-3)	4.0(-4)	5.0(-6)
Cs-Rb	7.7(-1)	5.0(-1)	3.4(-1)	9.3(-2)	8.1(-2)	6.2(-2)	9.0(-3)	1.0(-3)	1.0(-5)
Te	7.5(-1)	1.0(-1)	3.8(-1)	4.4(-2)	6.4(-2)	4.9(-2)	7.0(-3)	1.0(-3)	1.0(-5)
Ba-Sr	8.6(-2)	6.0(-2)	3.7(-2)	1.1(-2)	9.2(-3)	7.1(-3)	1.0(-3)	1.0(-4)	1.0(-6)
Ru	6.1(-2)	2.0(-2)	2.9(-2)	5.0(-3)	5.6(-3)	4.3(-3)	6.0(-4)	7.0(-5)	1.0(-[illegible])
La	9.8(-3)	2.0(-3)	4.9(-3)	6.6(-4)	8.6(-4)	6.6(-4)	9.0(-5)	1.0(-5)	2.0(-7)

Table III.B[illegible] (Continued)

INDIAN POINT RELEASE CATEGORY FREQUENCIES

Release Category	Unit 2						Unit 3					
	Before Fix			After Fix			Before Fix			After Fix		
	INT#	LOSP#	RD#	INT	LOSP	RD	INT	LOSP	RD	INT	LOSP	RD
A	0.0	0.0	7(-7)	0.0	0.0	7(-7)	0.0	0.0	3.5(-8)	0.0	0.0	3.5(-8)
B	4.6(-7)	3.6(-9)	3.2(-8)	4.3(-7)	3.6(-9)	4.3(-9)	4.7(-7)	0.0	1.3(-9)	4.3(-7)	0.0	1.3(-9)
C	1.6(-4)	6.4(-6)	1.3(-4)	1.2(-5)	6.4(-6)	1.7(-5)	1.5(-[illegible]	6(-7)	4.8(-6)	1.2(-5)	6(-7)	4.8(-6)
D	1.0(-6)	2.0(-9)	0.0	1.0(-6)	2.0(-9)	0.0	1.0(-6)	0.0	0.0	1.0(-6)	0.0	0.0
E	1.6(-7)	0.0	0.0	1.6(-7)	0.0	0.0	1.0(-7)	0.0	0.0	1.0(-7)	0.0	0.0
F	6.2(-6)	6.8(-7)	1.6(-6)	4.4(-6)	6.8(-7)	2.2(-7)	7.8(-6)	9.8(-8)	9.6(-8)	6.1(-6)	9.8(-8)	9.6(-8)
G	6.4(-7)	3.6(-8)	3.2(-7)	2.7(-7)	3.6(-8)	4.3(-8)	6.6(-7)	4.5(-9)	1.3(-8)	3.3(-7)	4.5(-9)	1.3(-8)
H	2.4(-4)	1.2(-5)	1.6(-4)	5.1(-5)	1.2(-5)	2.2(-5)	2.3(-4)	1.4(-6)	6.2(-6)	6.3(-5)	1.4(-6)	6.2(-6)
I	2.3(-4)	1.7(-5)	1.9(-5)	2.0(-4)	1.7(-5)	2.6(-6)	2.7(-4)	2.5(-6)	1.7(-6)	2.5(-4)	2.5(-6)	1.7(-6)
		1(-3)			3.5(-4)			6.8(-4)			3.4(-4)	

#For definitions of INT, LOSP, and RD see Figure II.B.1

##1.0(0) = 1.0 x 10^0

**NF = no failure

†The release fractions for these categories are higher than equivalent-category release fractions used in the RSS (WASH-1400)

††The release time for these categories can be arbitary; that is, the (risk) results are insensitive to variations in the release time.

distributions for source terms which assign a high value to source term estimates varying by more than an order of magnitude simply demonstrate the substantial uncertainties associated with these estimates. The radionuclide release terms (source terms) used by the staff are appropriate at the present time.

Q.15 Before continuing with step 6, an analysis of the reduction or elimination of various containment failure modes due to the presence of mitigation features, please explain the purpose of a mitigation feature.

A.15 The purpose of a mitigation feature is to mitigate the consequences of severe accidents, accidents that are beyond the design basis of nuclear reactor containment buildings by reducing or eliminating one or several of the containment building failure modes discussed in this testimony. It is, however, important to stress that the existing containment buildings adequately mitigate the consequences of a wide range of postulated accidents that are more severe than those considered in the original design of the building. A new mitigation feature, combined with an existing containment building design, will mitigate the consequences of an even wider range of severe accidents.

Q.16 Please discuss how the staff determines the safety benefit afforded by a particular mitigation feature.

A.16 The safety benefit of a mitigation feature can be determined qualitatively by assessing its capability to eliminate or reduce the effect of a particular containment building failure mode. This process can proceed without resorting to probabilistic risk analysis. However, it is our opinion that we should quantify the safety benefit of a mitigation feature by using the approach described in this testimony. This approach allows for a quantitative measure of safety benefit by determining the risk reduction resulting from such a feature, thus providing for a direct link with effects on the public.

Q.17 Please explain how you assess the limitations and drawbacks of mitigation features.

A.17 A practical engineered safety system will have an inherent unreliability and potential negative characteristics that must be taken into account in any assessment of its safety benefit. Thus practical engineering conceptual designs are considered that meet certain functional requirements and design criteria. Based on the conceptual designs, unreliability can be estimated. In addition to unrealiability, it is very important to consider potential negative characteristics of a mitigation feature. In a probabilistic risk analysis context these negative features can be considered "attendant risks," that is, new risks that are introduced by the character of the feature itself. For example, a core-retention system that requires a flooded cavity introduces an attendent risk of increased potential for slow overpressurization failure of the containment building. The implementation of unreliability and negative characteristics into a risk analysis framework is described further in "Step-6" of the testimony.

Q.18 Please continue with Step 6, analysis of mitigation features.

A.18 Step 6 All steps discussed to this point represent the Indian Point facility as built. In step 6 we will consider the impact of mitigation features on radiological releases to the environment by following the methodology described in steps 2, 3, 4 and 5 of this testimony. We will first consider ideal mitigation features and then realistic mitigation features. Ideal features prevent the following containment failure modes by meeting the following requirements:

(1) For combustible gas control (preventing hydrogen burn [γ] failure mode), either (a) provide for the controlled burning of an amount of combustible gas sufficient to render the containment building inert by oxygen depletion in such a way that thermal or pressure loadings from controlled burning do not cause vital equipment or the containment building to fail; or (b) render the containment atmosphere inert either before or after accident initiation in such a way that the

containment building does not fail from pressure loadings contributed by this activity.

(2) For control of gradual overpressurization of the containment building (preventing overpressurization [δ] failure mode), provide a reliable means to remove the energy causing overpressurization so that the containment building failure pressure is not exceeded and so that the containment building pressure is brought below the design pressure within about 12 hours of initiation of the control measures. The basis for the 12-hour period recommended is the need to limit the initial leakage that would occur at pressures in excess of those the building was designed to withstand.

(3) For control of basemat penetration (for preventing basemat penetration [ε] failure mode), assure that interactions between the core and concrete are limited by establishing a coolable debris bed in the reactor cavity.

Based on the above requirements, we choose, among the options available, the following features for further consideration in our risk analysis:

(1) To control combustible gases: an ignition system to control burning using glow plug igniters.

(2) To control building overpressurization: a passive containment building heat removal system, such as heat pipes.

(3) For prevention of basemat penetration: a system to flood the reactor cavity.

As long as these features function ideally as designed, are 100% reliable, and do not themselves introduce any negative characteristics, the impact of these features on releases of radioactive materials from the containment building can be determined using the containment event trees by assigning a split fraction for failure of zero to those accident failure modes for which the mitigation feature is designed. Here we consider all three mitigation features as a single mitigation strategy and assign split fractions of zero to the δ, γ and ε failure modes.

The nine release categories given in Table III.B.3 were determined to be sufficiently representative of the full spectrum of releases that only the probabilities of releases for the nine categories had to be changed. These new probabilities are listed in Table III.B.4, together with the case discussed above for Indian Point Units 2 and 3 before mitigation.

No mitigation feature functions ideally as designed all of the time. In addition to unreliability, negative characteristics may be introduced when mitigation features are incorporated into a design. A realistic case was run in which negative features and unreliability were considered for controlled hydrogen burning, using glow plugs; for overpressurization protection, using a passive containment heat removal system; and, for basemat penetration, using a continually reflooded reactor cavity. The negative features and unreliability for this migitation strategy are discussed in Appendix B to this testimony. Again, as with the ideal cases, split fractions at the various branch points in the containment event trees are changed to reflect these unreliable and negative characteristics, and a new set of probabilities for the nine release categories is determined. These are also shown in Table III.B.4. Figure III.B-4, 5, 6, and 7 together provide the complete set of containment event trees for Indian Point as is (probability column "a"), with ideal mitigation (probability column "b"), and with realistic mitigation (probability column "c").

Step 7 With the impact of the mitigation features addressed, the risk-reduction analysis can proceed by use of the CRAC consequence analyses. This is the subject of Section III.C, "Staff assessment of accident consequences."

Q.19 Please summarize the key aspects of your analysis and assessment.

A.19 The analysis and assessment performed in this section can be summarized with the aid of Figure III.B.8. Coremelt plant damage states and their

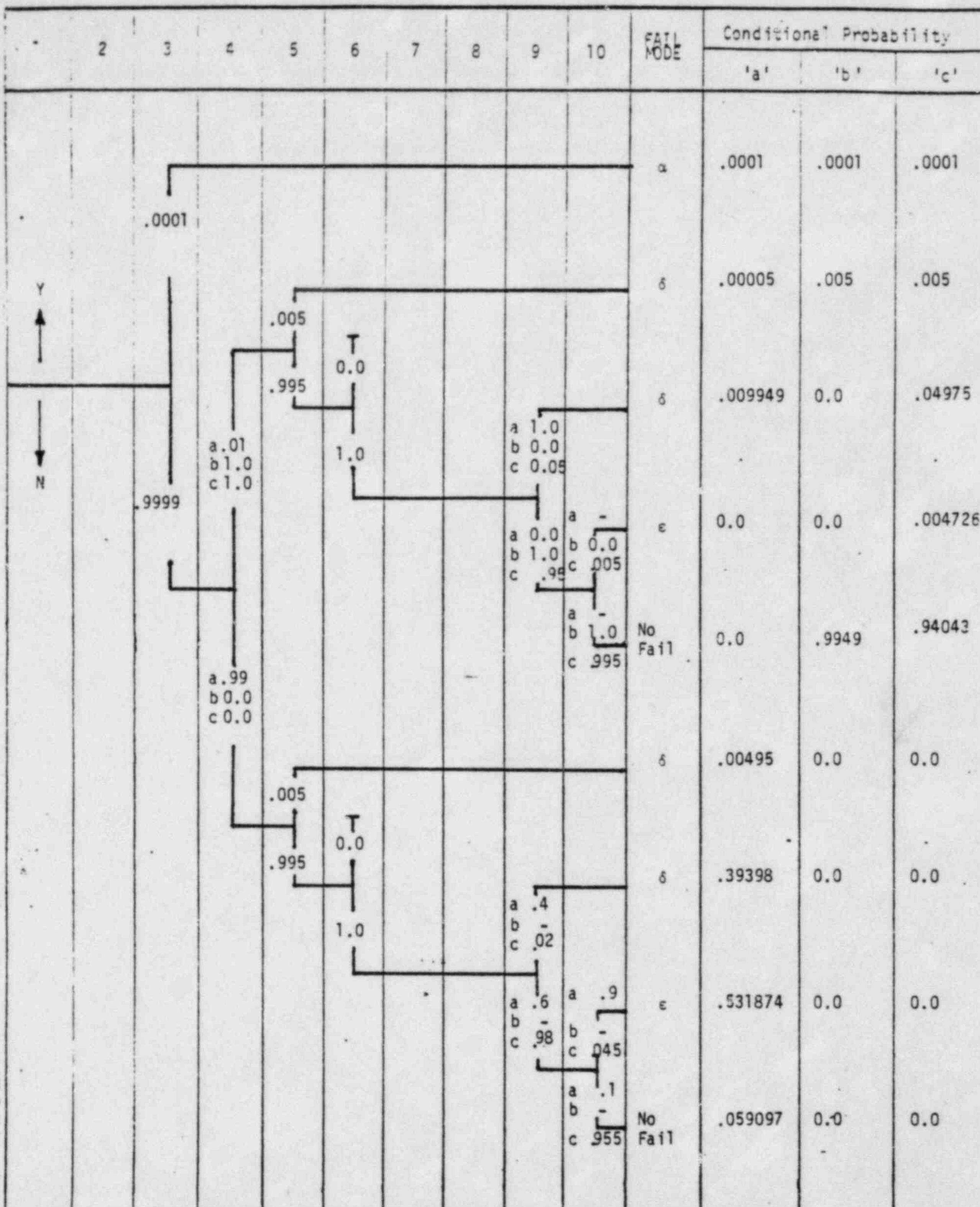
Table III.B.4 Probability assigned to each release category

Release Category		Unit 2 1	Unit 2 2	Unit 2 3	Unit 3 1	Unit 3 2	Unit 3 3
A	INT & LOSP	-	-	-	-	-	-
	RD	7.0(-7)	7.0(-7)	7.0(-7)	3.5(-8)	3.5(-8)	3.5(-8)
B	INT & LOSP	4.3(-7)	4.3(-7)	4.3(-7)	4.3(-7)	4.3(-7)	4.3(-7)
	RD	4.3(-9)	4.3(-9)	4.3(-9)	1.3(-9)	1.3(-9)	1.3(-9)
C	INT & LOSP	1.8(-5)	2.0(-6)	4.0(-6)	1.3(-5)	2.0(-6)	3.3(-6)
	RD	1.7(-5)	0.0	2.1(-6)	4.8(-6)	0.0	6.0(-7)
D	INT & LOSP	1.0(-6)	0.0	9.0(-8)	1.0(-6)	0.0	1.0(-7)
	RD	-	0.0	0.0	-	0.0	0.0
E	INT & LOSP	1.6(-7)	0.0	8.1(-9)	1.0(-7)	0.0	6.5(-9)
	RD	-	0.0	0.0	-	0.0	0.0
F	INT & LOSP	5.0(-6)	2.2(-7)	3.2(-7)	6.1(-6)	1.6(-7)	2.8(-7)
	RD	2.2(-7)	2.2(-7)	2.2(-7)	9.6(-8)	6.0(-8)	6.0(-8)
G	INT & LOSP	3.0(-7)	3.0(-7)	3.0(-7)	3.3(-7)	3.3(-7)	3.3(-7)
	RD	4.3(-8)	4.3(-8)	4.3(-8)	1.3(-8)	1.3(-8)	1.3(-8)
H	INT & LOSP	6.3(-5)	0.0	1.5(-6)	6.4(-5)	0.0	1.6(-6)
	RD	2.2(-5)	0.0	2.0(-7)	6.2(-6)	0.0	6.3(-8)
I	INT & LOSP	2.2(-4)	3.0(-4)	3.0(-4)	2.5(-4)	3.3(-4)	3.3(-4)
	RD	2.6(-6)	4.3(-5)	4.0(-5)	1.7(-6)	1.3(-5)	1.2(-5)

[1]Probabilities Before Mitigation
[2]Probabilities with Ideal Mitigation Strategy
[3]Probabilities with Realistic Strategy

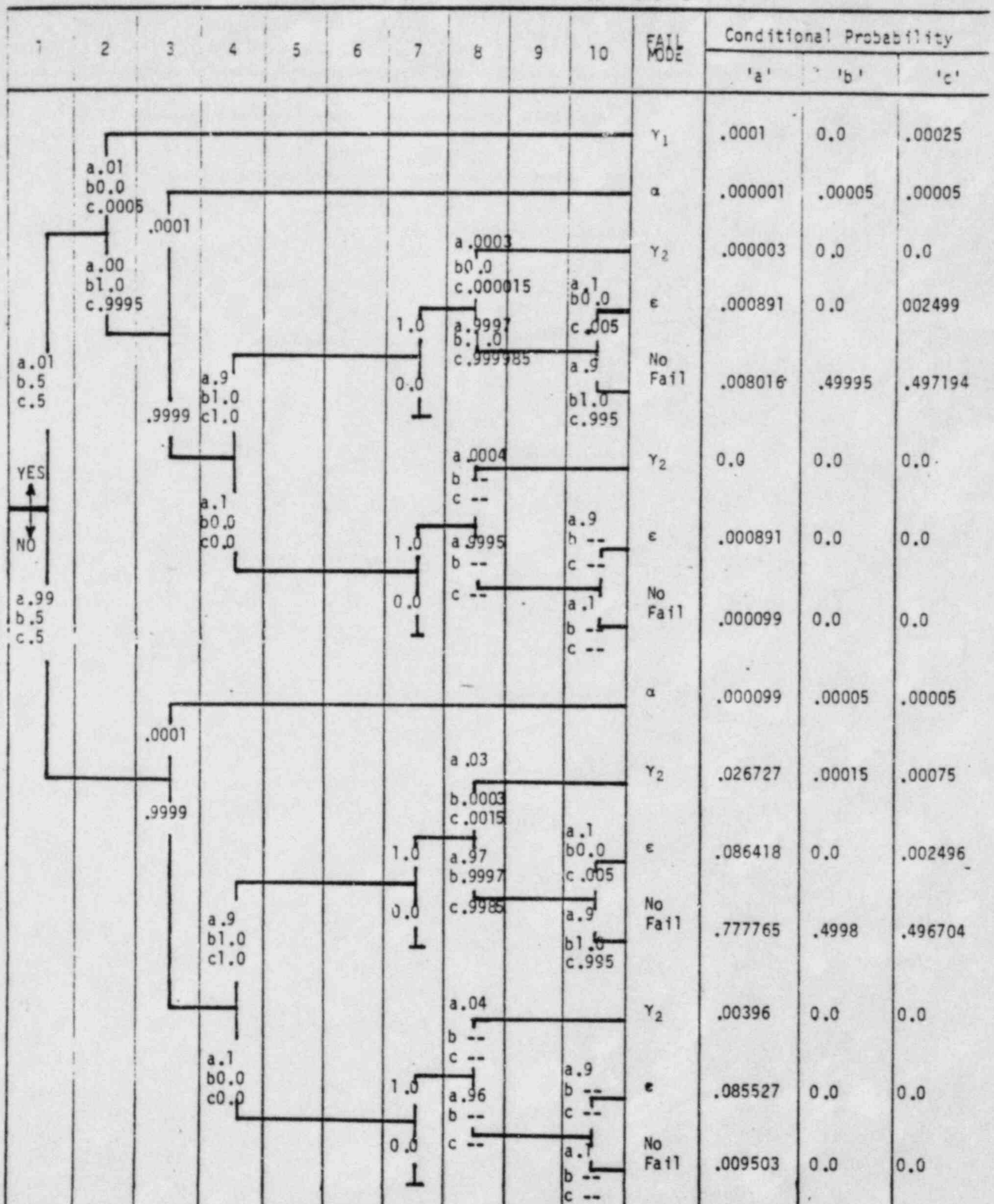
FIGURE III.B-4

ACCIDENT SEQUENCE E INDIAN POINT UNIT 2



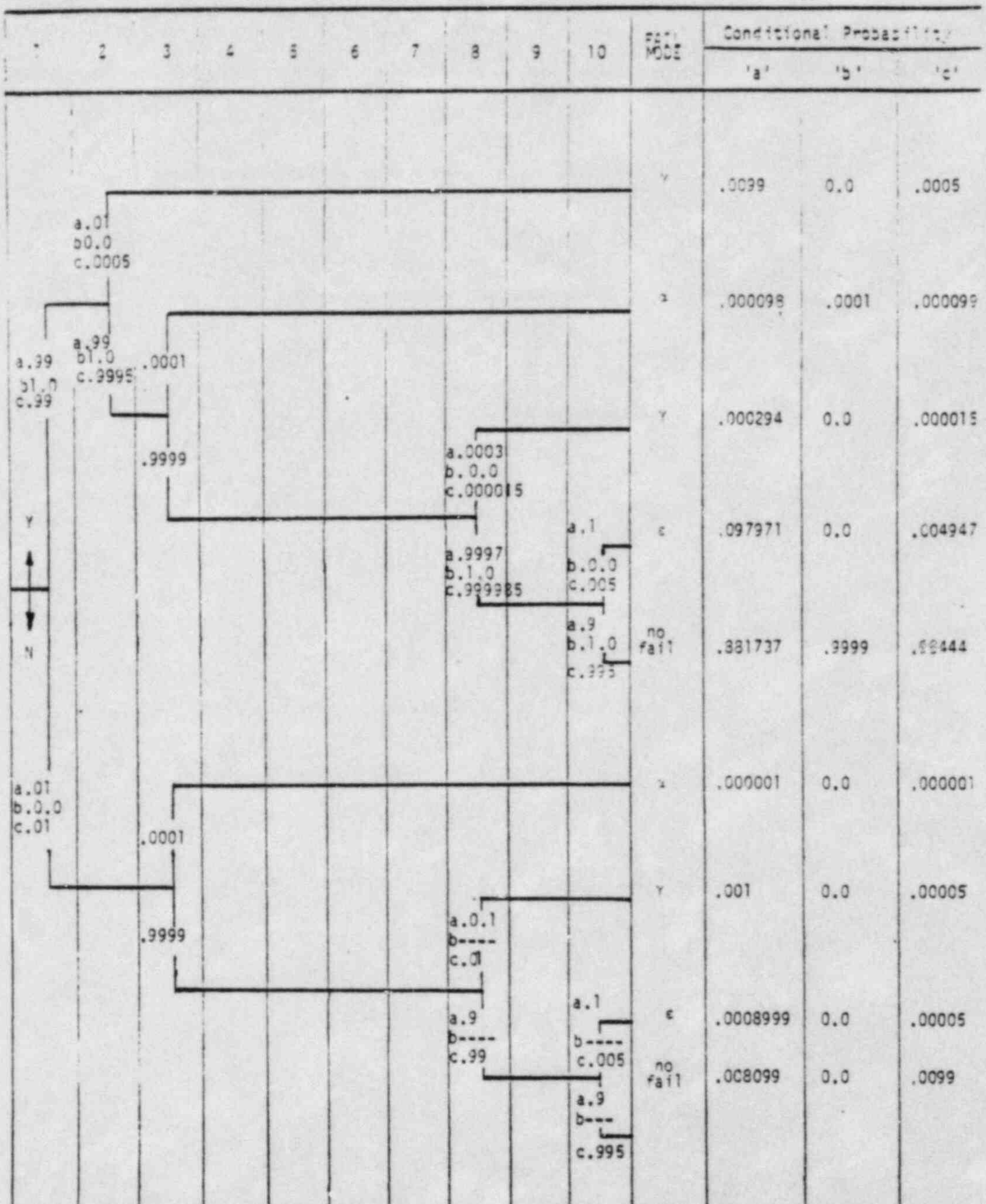
FAILURE MODE	'a' BEFORE FIX		'b' IDEAL STRATEGY		'c' REALISTIC STRATEGY	
	P_c	$P(10^{-5})$	P_c	$P(10^{-5})$	P_c	$P(10^{-5})$
α (CATEGORY B)	.0001		.0001		.0001	
δ_1 (CATEGORY F)	.005		.005		.005	
δ_2 (CATEGORY C)	.40393		.0.0		.04975	
ε (CATEGORY H)	.53187		0.0		.004726	
NO FAIL (CATEGORY I)	.05910		.9949		.94043	
TOTALS	1.0		1.0		1.0	

FIGURE III.B-5
ACCIDENT SEQUENCE EFC INDIAN POINT UNIT 2



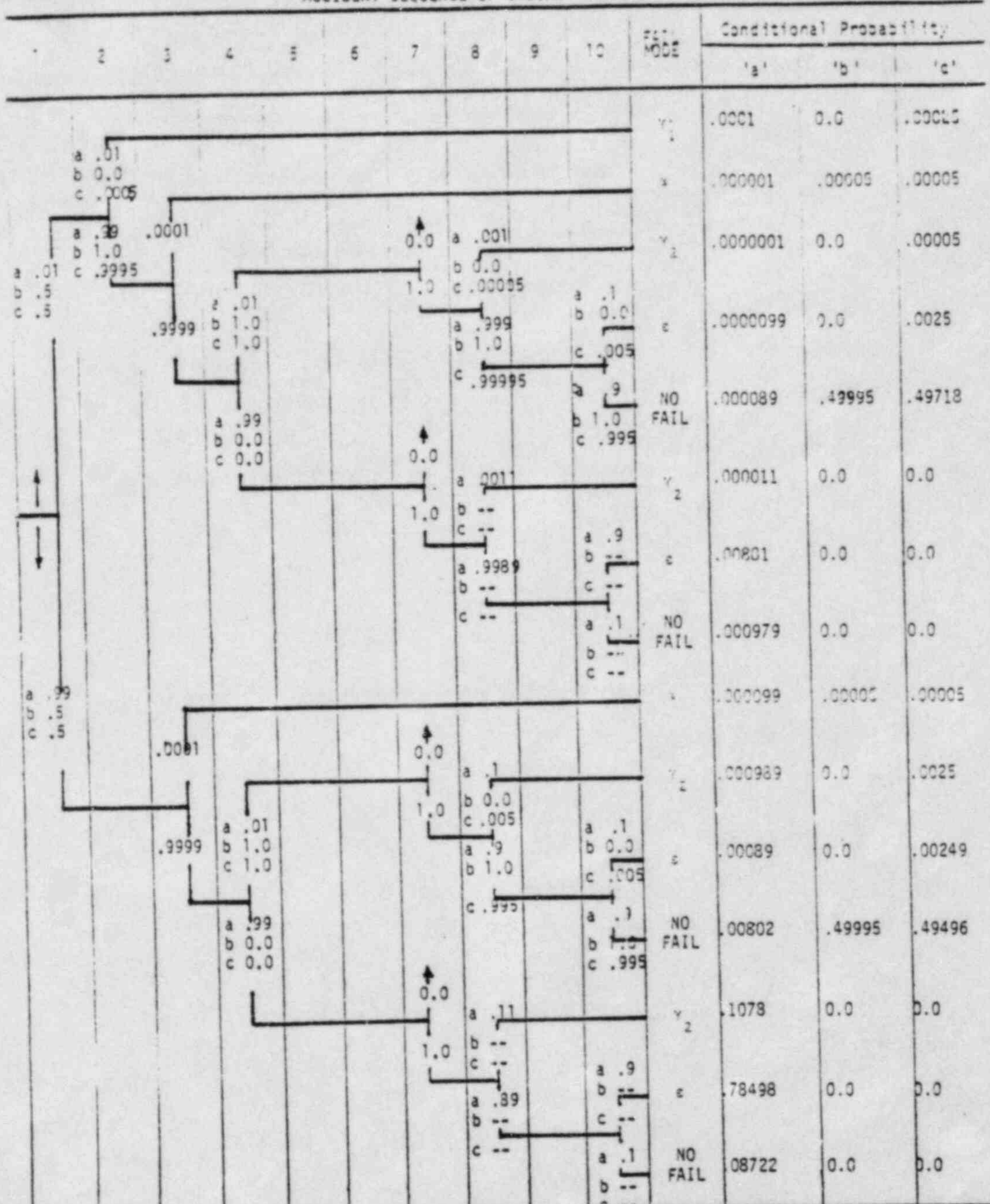
FAILURE MODE	'a' BEFORE FIX		'b' IDEAL STRATEGY		'c' REALISTIC STRATEGY	
	P_c	$P(10^{-5})$	P_c	$P(10^{-5})$	P_c	$P(10^{-5})$
α (CATEGORY B)	.0001		.0001		.0001	
γ_1 (CATEGORY D)	.0001		0.0		.00025	
γ_2 (CATEGORY F)	.03069		.00015		.00075	
ϵ (CATEGORY H)	.17373		0.0		.004995	
NO FAIL (CATEGORY I)	.79538		.99975		.993898	
TOTALS	1.0		1.0		1.0	

FIGURE III.B-6
ACCIDENT SEQUENCE LF INDIAN POINT UNIT 2



FAILURE MODE	'a' BEFORE FIX		'b' IDEAL STRATEGY		'c' REALISTIC STRATEGY	
	'a' P_c	$P(10^{-5})$	'b' P_c	$P(10^{-5})$	'c' P_c	$P(10^{-5})$
α (CATEGORY B)	.000099		.0001		.000099	
γ_1 (CATEGORY D)	.0099		0.0		.0005	
γ_2 (CATEGORY E)	.001294		0.0		.000065	
ϵ (CATEGORY H)	.098871		0.0		.004997	
NO FAIL (CATEGORY I)	.889836		.9999		.994844	
TOTALS	1.0		1.0		1.0	

FIGURE III.B - 7
ACCIDENT SEQUENCE EF INDIAN POINT UNIT 2



FAILURE MODE	'a' BEFORE FIX		'b' IDEAL STRATEGY		'c' REALISTIC STRATEGY	
	P_c	$P(10^{-5})$	P_c	$P(10^{-5})$	P_c	$P(10^{-5})$
α (CATEGORY B)	.0001		.0001		.00010	
γ_1 (CATEGORY D)	.0001		0.0		.00025	
γ_2 (CATEGORY E)	.1088		0.0		.00253	
ϵ (CATEGORY H)	.79469		0.0		.005	
NO FAIL (CATEGORY I)	.09631		.9999		.99212	
TOTALS	1.0		1.00		1.000	

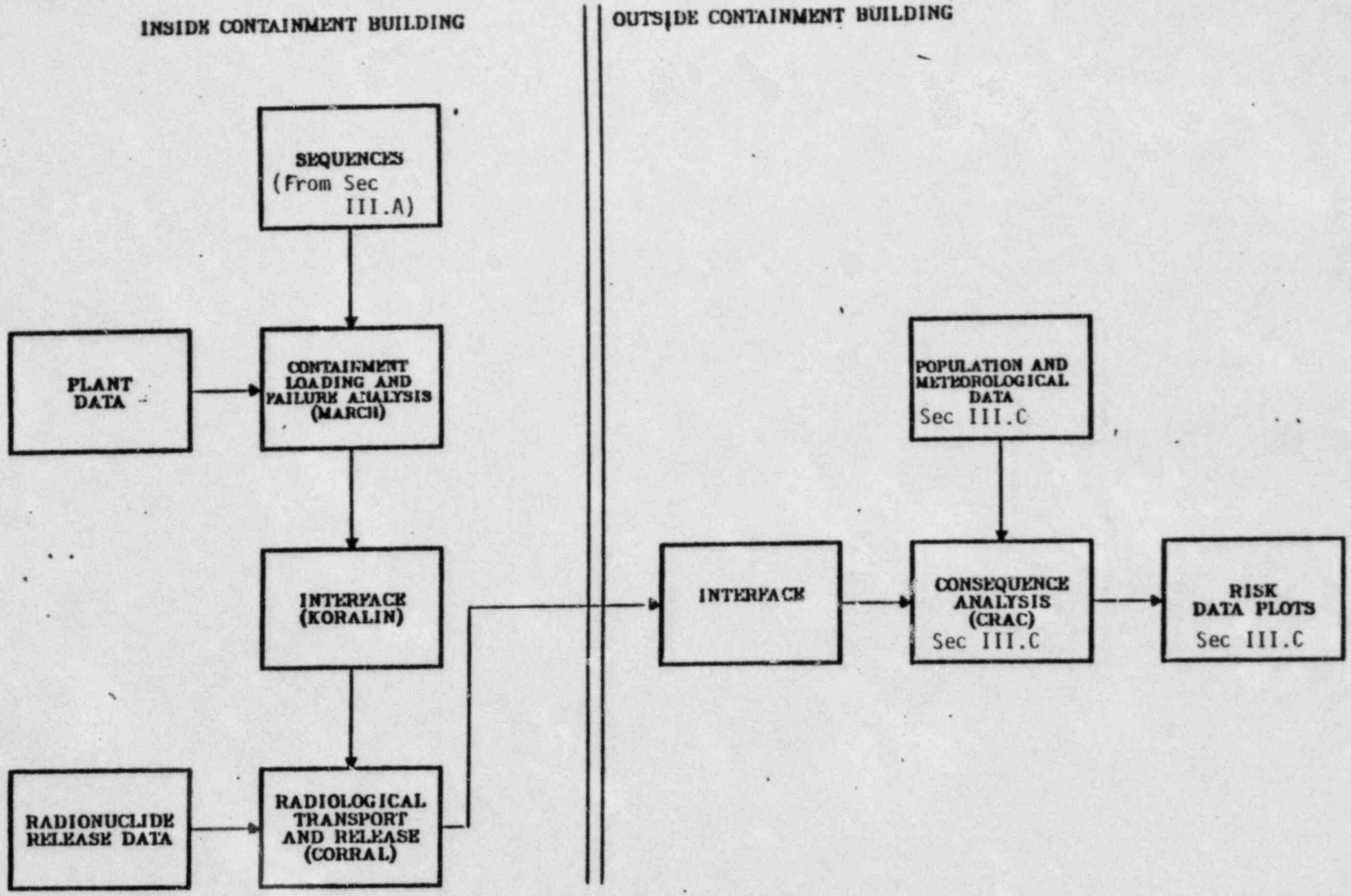


Figure III.B-8: **A flow schematic for risk analysis** associated probability were provided from Section III.A. Using these damage states, Indian Point Units 2 and 3 plant data, and radionuclide release data, analyses and assessments were performed to determine how and when these containment buildings fail, the amount of radionuclide releases upon failure, and changes in these releases due to the presence of mitigation features. The analysis yielded 9 release categories, that represent the full spectrum of releases attributable to the various containment building failure modes. These categories in order of decreasing cesium release fraction, that is decreasing consequence impact, are:

Release Category	Failure Mode
A	Large seismic event β^* (containment collapse)
B	Event V and all α (alpha) failure modes
C	All long-term δ (delta) overpressurizations and SGTR event
D	All early γ (gamma) hydrogen burns (no sprays)
E	All late γ (gamma) hydrogen burns (no sprays)
F	All early γ (gamma) hydrogen burns (with sprays)
G	All β (beta) failure modes (failure to isolate containment)
H	All ε (epsilon) basemat penetration modes
I	All conditions for which containment failure does not occur.

Just how the probabilities for the various containment failures are partitioned among the nine release categories is shown in Figure III.B.9 for Unit 2 and Figure III.B.10 for Unit 3. (These figures summarize the data

Containment Frequency of Release For Unit 2

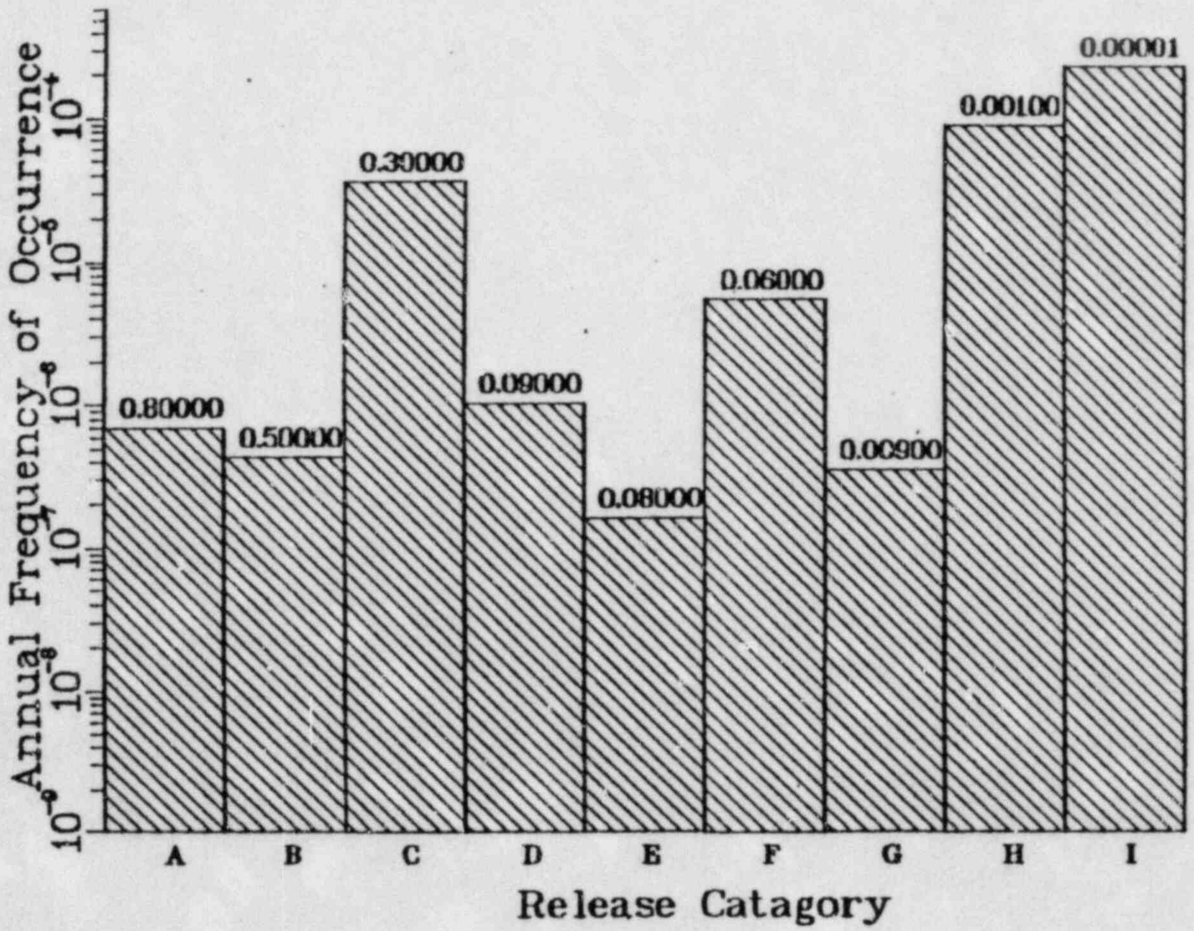


Figure III.B-9

Containment Frequency of Release For Unit 3

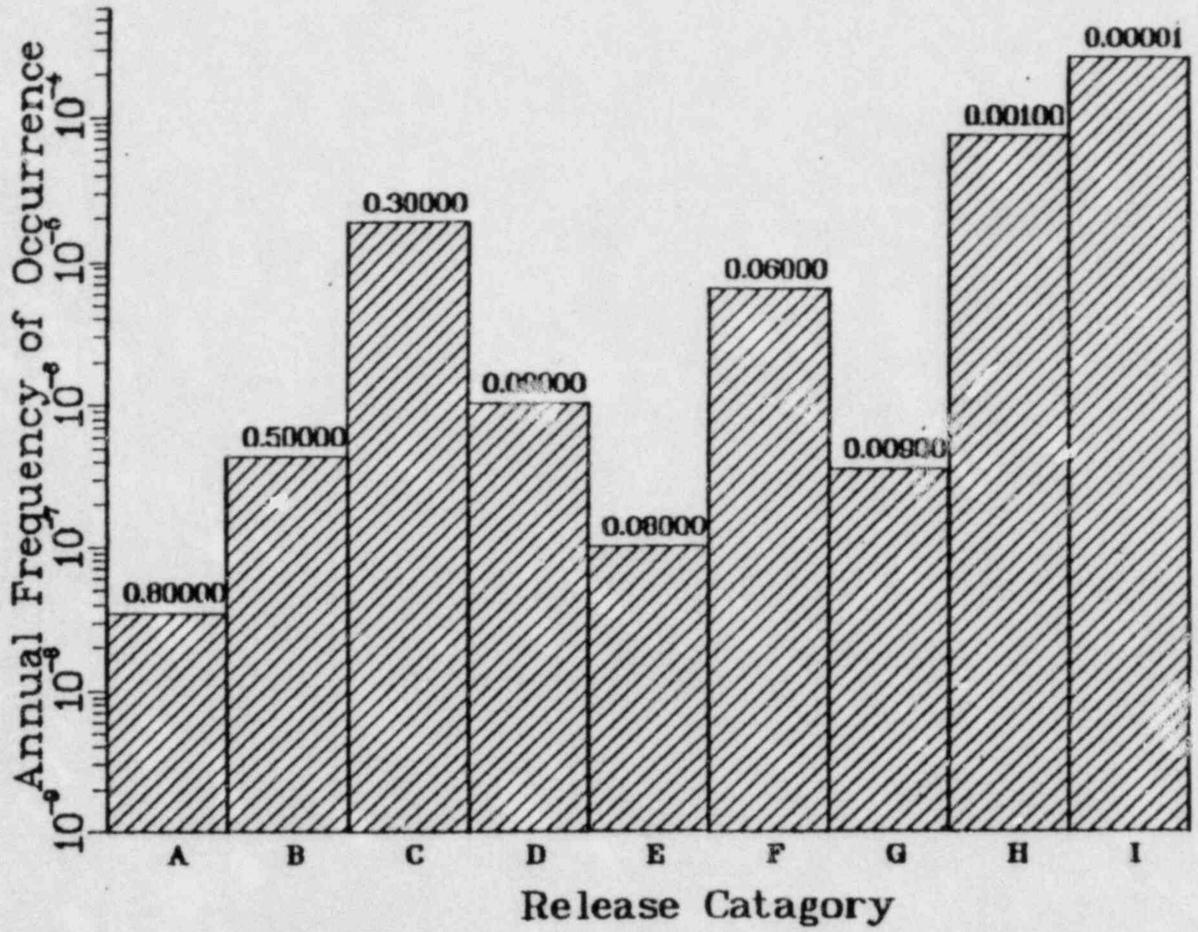


Figure III.B-10

in Table III.B.3.) In terms of probability of release, the dominant release categories for both units are C (for slow overpressurization), H (for basemat penetration), and I (for "no failure"). However, when the amount of radionuclide is considered, categories H and I become unimportant. The release categories that dominate risk then will be A, B, and C. The C category dominates due to the combination of high relative probability and large release; the A and B also are potentially major contributors due to their large release fractions.

Of the three dominant release categories, only C can be reduced or eliminated by mitigation features. The changes in this release category that can be anticipated with ideal and realistic mitigation strategies are shown in Table III.B-4 for both Units 2 and 3. The mitigation feature considered is a passive containment heat removal system (heat pipe system). The other components of the mitigation strategy, a distributed ignition system for hydrogen control, and a core retention system to prevent basemat penetration, change other release categories (categories D, E, and F for hydrogen control and H for core retention), but the changes are not as significant as the changes resulting from overpressurization control.

The impact of the mitigation strategies on the radionuclide releases from the containment are graphically displayed in Figures III.B.11 and 12, for Units 2 and 3, respectively. The amount of cesium (Cs) released is used as an index of the severity of the release. The probability of exceeding a release is plotted as a function of the amount released. Thus, since the probability of exceeding a Cs release fraction (for unit 2) of 0.4 (40%) is zero for all the categories except A; the right-hand element of the graph shows the contribution from release category A--the most severe. Note that it is not reduced by the mitigation strategy. Based on the data in Tables III.B.3 and III.B.4, the other release categories are added, yielding the graph shown. Note that the left-most element of the graph is the sum of all probabilities and thus the total probability of coremelt. It is clear then that there is substantial reduction in releases of cesium when the ideal mitigation strategy is in place. It is also clear that there is a substantial loss in that reduction for a realistic strategy which considers unreliability and negative aspects of this strategy.

INDIAN POINT 2 RELEASE PROFILE

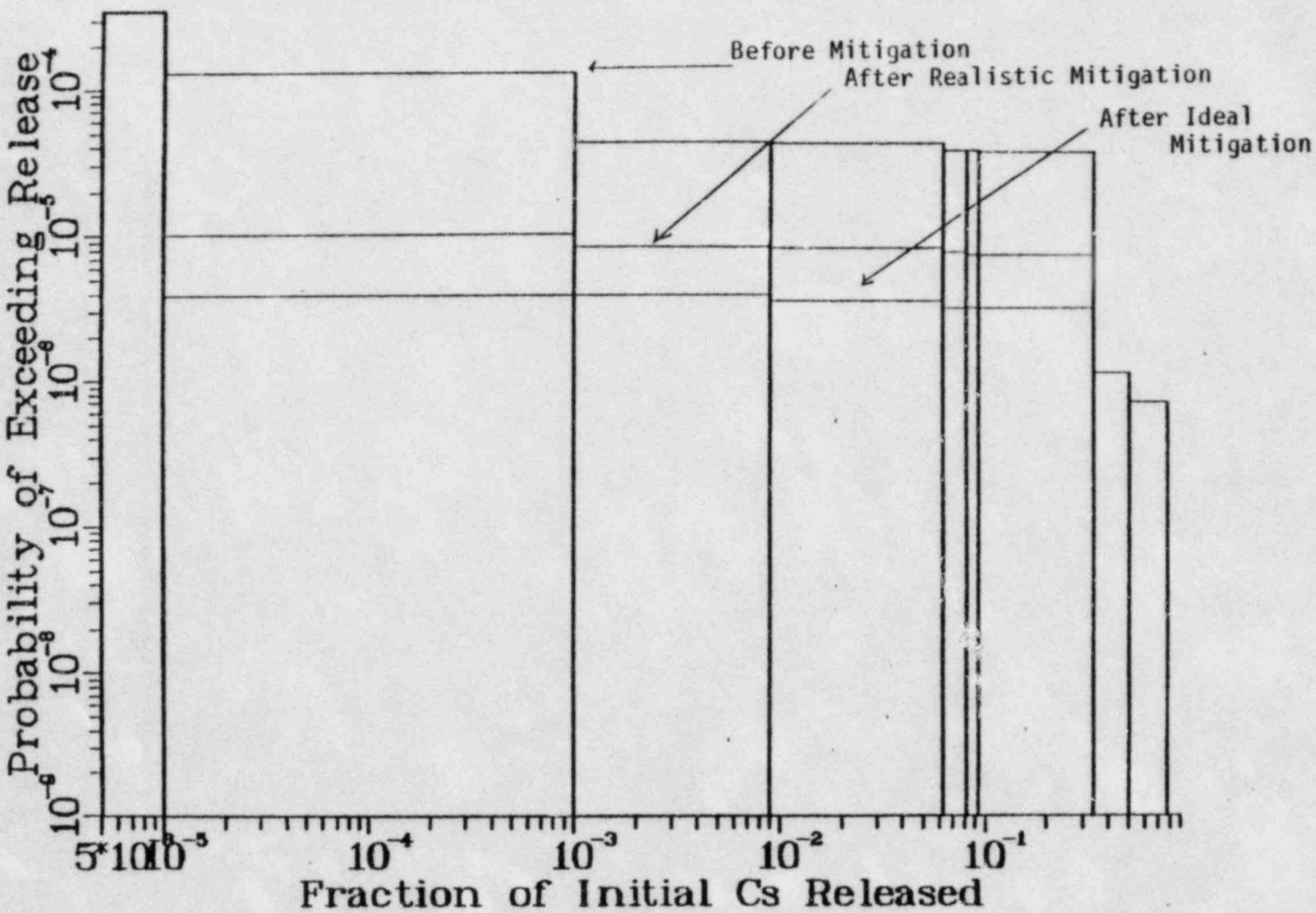


Figure III.B-11

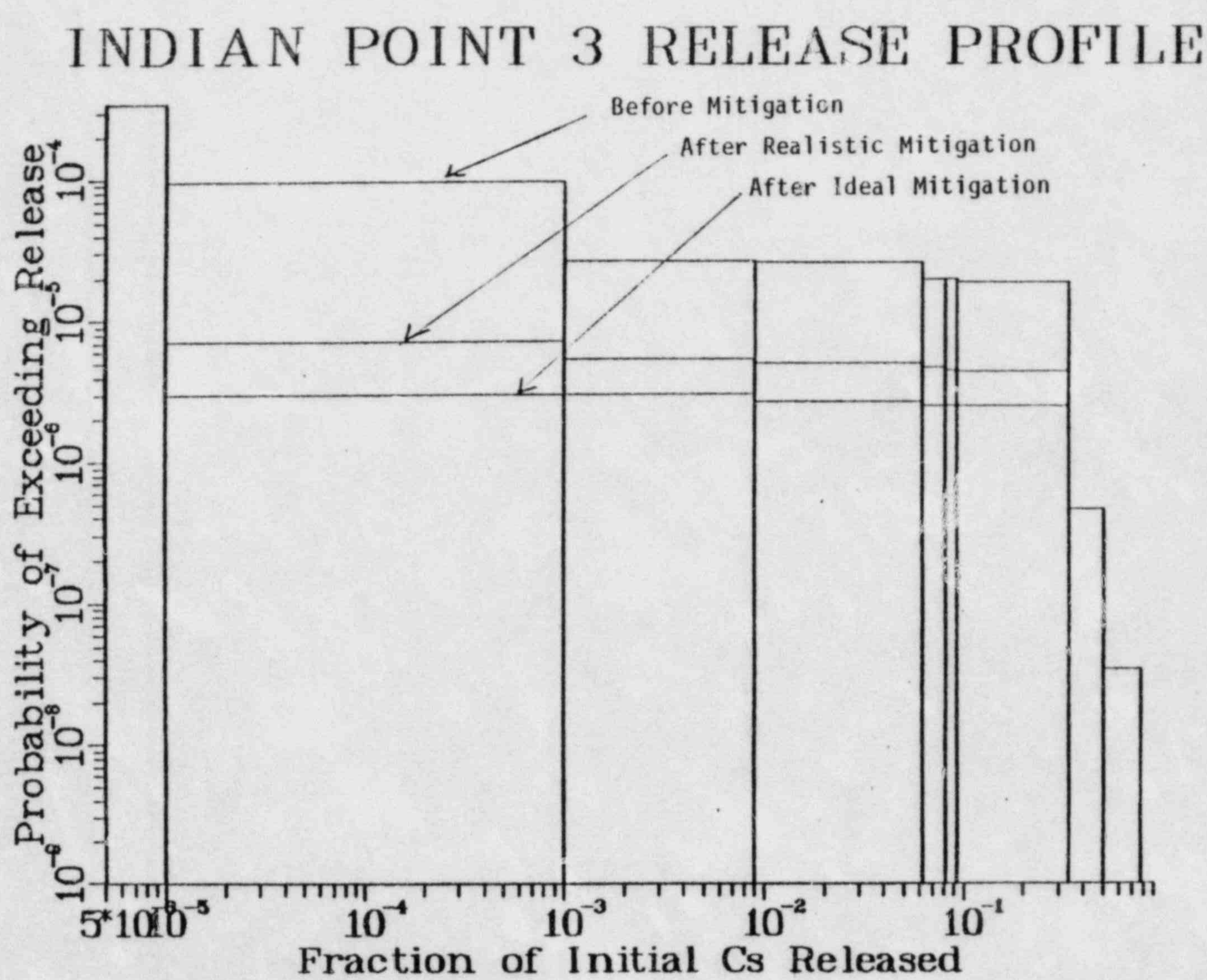


Figure III.B-12

Although this depiction of the safety benefits of mitigation is useful, a complete consequence analysis must be performed in order to have more definitive results. These consequence analyses are the subject of the testimony of Dr. Acharya. The results of those analyses that have a direct bearing on the risk reduction from mitigation are summarized here, however, in order to complete the evaluation of mitigation features.

Estimators of the safety benefit (risk reduction) from the mitigation features have been determined using the methodology described in Section III.C. Ways of making these estimates can take several different forms:

(a) by plotting CCDF (complementary cumulative distribution function) curves comparing the societal risks before and after incorporating mitigation strategies for various risk measures (e.g., early fatalities, delayed cancer fatalities);

(b) by plotting curves of individual risks as a function of distance from the facility, again comparing risks before to those after incorporating mitigation strategies; and

(c) by comparing the numerical values obtained by integrating the CCDF curves which represent the risks before and after mitigation strategies have been implemented. These numerical values represent the values expected for societal risk. As with the CCDF curves themselves, the comparison can be made for a variety of risk measures.

Here we choose to use form (c). The numerical values are determined by multiplying the conditional mean values for societal consequences for each release category (as listed in Table III.C.5 for consequence categories 1 [early fatalities] and 5 [delayed cancer fatalities]) times the probability for each release category (as listed in Table III.B.4) and summing to determine the total risk numerical values. This is done for the three cases under consideration here, namely before mitigation, with ideal mitigation and with realisitic mitigation. A summary of these expectation values for Units 2 and 3 follow:

Table III.B.5
Indian Point Unit 2

	Delayed Cancer Fatalities (per Reactor-Yr.)	Early Fatalities (per Reactor-Yr.)
Before Mitigation	1.7 (-1)*	1.5 (-2)
After Mitigation (realistic features)	3.4 (-2)	7.7 (-3)
After Mitigation (ideal features)	1.6 (-2)	6.6 (-3)

*1.7(-1) = 1.7×10^{-1}

Table III.B.6
Indian Point Unit 3

	Delayed Cancer Fatalities (per Reactor-Yr.)	Early Fatalities (per Reactor-Yr.)
Before Mitigation	9.1 (-2)	3.8 (-3)
After Mitigation (realistic features)	1.9 (-2)	1.3 (-3)
After Mitigation (ideal features)	1.0 (-2)	9.5 (-4)

We formulated a quantitative risk comparison by using the above values and the following relationship:

$$\frac{[\text{Expectation Value Before} - \text{Expectation Value After}]}{\text{Expectation Value Before}} \times 100$$

that is, the risk reduction normalized to the initial risk. The normalization yields a risk reduction percentage measure independent of absolute risk. Using this formulation, we have:

Table III.B.7 Indian Point Unit 2

	Risk Reduction (Delayed Cancers)	Risk Reduction (Early Fatalities)
Mitigation Strategy (realistic features)	80%	50%
Mitigation Strategy (ideal)	91%	56%

Table III.B.8 Indian Point Unit 3

	Risk Reduction (Delayed Cancers)	Risk Reduction (Early Fatalities)
Mitigation Strategy (realistic features)	80%	66%
Mitigation Strategy (ideal)	90%	75%

Figures III.B.5, 6, 7 and 8 display these risk reduction results as segments of total risk.

It is apparent that the potential impact of mitigation strategies on delayed cancer fatalities is significant.

The potential risk reduction on "early fatalities" is reduced; that is, the risk reduction percentage is smaller. One reason for the different risk reduction values when considering latent versus early fatalities can be seen by noting the release categories in Tables III.B-3 and III.B-4. Early fatalities, having a localized threshhold impact, are controlled primarily by the highest "release fraction" release category, in this case categories A, B and C. (Note Table III.B-3.) The probability for the A and B release categories remains constant when considering mitigation feature strategy options (note Table III.B-5), since none of the strategies is considered a "fix" for the large seismic event, the event V, or the steam-explosion containment failure modes. Put another way, early fatalities result only from the large release-fraction categories, categories where, with the exception of release category "C," mitigation features have little or no impact.

Q.20 Please comment on the impact of uncertainties on your results.

A.20 As described in several parts of this testimony, there are significant uncertainties and unknowns in areas of phenomenology, the accident progression and containment failure characteristics. In order to get a better idea of the impact of these uncertainties and unknowns on risk values, we have performed a parametric analysis by varying key parameters which have large unknowns and uncertainties and noting the effect of these variations on the release categories and on the actual risk values. The change in the risk values reported here are determined from risk analyses essentially identical to those described in Sarbaswar Acharya's testimony which describes, in detail, the determination of public risk given the radionuclide release data generated in this testimony.

Uncertainties in the following areas are considered here:

1) Uncertainty in the ability of hydrogen burns to fail the containment building

2) Uncertainty in failure of the containment building by gradual overpressurization

3) Uncertainty in the ability of a flooded cavity to establish a coolable debris bed and therefore prevent basemat penetration

4) Uncertainty in whether or not the containment building fan coolers can perform their function under the adverse environmental conditions of a severe accident

5) Uncertainties in the performance (reliability) of the heat-pipe mitigation feature

The uncertainty assessment presented here is in the form of percentage changes from the original risk for two risk measures; early fatalities and late fatalities. Percentage changes are indicated for both Indian Point 2 and Indian Point 3.

1) If we increase the probability of failing the containment building by hydrogen burns by an order of magnitude (e.g., a split fraction change from 0.03 to 0.30), the percentage change in risk values are:

% Change	I.P. #2	I.P. #3
early fatalities	neg	neg.
late fatalities	40%	100%

On the other hand, if we reduced the probability by an order of magnitude Ie.g., a split fraction change from 0.03 to 0.003), the percentage change in risk values are:

% Change	I.P. #2	I.P. #3
early fatalities	neg.	neg.
late fatalities	-4%	-10%

We conclude that the effect on early fatalities of large parametric variations on the hydrogen-burn failure mode is small and negligible. This is not surprising since the contributions to early fatalities from the hydrogen-burn release categories (D, E, F) are small.

On the other hand, late fatalities are affected by large variations in the hydrogen burn failure modes, particularly for Unit 3. However, it should be noted that a change in the failure mode by an order of magnitude only changes risk by a maximum factor of 2.

2) If we increase the probability of failing the containment building by gradual overpressurization by 75% from a split fraction of 0.4 to 0.7, the percentage change in risk values are

% Change	I.P. #2	I.P. #3
early fatalities	45%	55%
late fatalities	55%	60%

These calculations show the sensitivity of parametric variations related to overpressurization failure on the final risk values. Thus the phenomenological and containment building failure pressure uncertainties associated with this failure mode are relatively important. But as important as they are, their effects are risk increases that are less than a factor of two.

3) If we assume that the probability of basemat penetration is 50% for the flooded cavity case instead of the 10% that we determined in the analysis, that is we assumed a more pessimistic position regarding the establishment of a coolable debris bed in the flooded cavity, the analysis yields

% Change	I.P. #2	I.P. #3
early fatalities	neg.	neg.
late fatalities	3%	5%

Thus, the changes in risk are all less than 10%, pointing out the insensitivity of the overall risk due to whether or not the basemat is penetrated.

4) If the fan coolers fail due to environmental effects resulting from the core/water/concrete interactions, then the plant damage states that had characteristics of an "EF" and a "LF" will both look more like "E" and "L" damage states with overpressurization failure dominating. An analysis was performed assuming that the cooling fans failed for the "LF" damage state 25% of the time* with the following results:

% Change	I.P. #2	I.P. #3
early fatalities	neg.	neg.
late fatalities	32%	63%

For early fatalities the failure of the fan coolers had little effect due to the small contribution to this risk measure from the EF and LF plant damage states. For late fatalities, on the other hand, the increases in risk are larger because of the prominence played by these damage states.

Although these parametric calculations alert us to the potential importance of the fan coolers, we are also aware that the design of

*A similar analysis was performed for the "EF" damage state. For both units and both risk measures, the changes in risk were negligible.

the fan coolers is such that the potential for failure from environmental conditions is small.

5) Of the three mitigation features considered in this study, the most important is the feature that prevents gradual overpressurization failure, namely the passive containment heat removal system.

The risk reduction provided by this system is given below:

% Change	I.P. #2	I.P. #3
early fatalities	57%	75%
late fatalities	83%	73%

This above assumes that the system functions ideally as designed at 100 reliability and does not introduce any negative characteristics. The unreliability of this system suggested in Appendix B is 5%. At this 5% unreliability, the following risk reduction is obtained:

% Change	I.P. #2	I.P. #3
early fatalities	50%	66%
late fatalities	73%	64%

If the unreliability is 10%, the following risk reduction is obtained.

% Change	I.P. #2	I.P. #3
early fatalities	42%	56%
late fatalities	63%	55%

The above parametric study indicates the benefit to be gained from improving the reliability of the passive heat removal mitigation feature.

In conclusion, this parametric study showed for the most part that parameters associated with major uncertainties could be varied by large amounts with little effect on the final results. The two exceptions were the parametric analyses of the overpressurization and hydrogen burn failure modes, but even then the variations yielded changes in risk of a factor of two or less, and then only for the latent fatalities risk measure. Although we are not in a position to say that this parametric exercise encompasses the key uncertainties, we believe the results are indicative of the variation in results based on the major uncertainties.

Q.21 Please compare the staff containment assessment with the IPPSS assessment considering first the question of overpressurization from hydrogen burning.

A.21 The staff assessment of the potential for hydrogen burns to fail the containment building (γ-failure mode) differs from the IPPSS assessment in two key areas: (1) the amount of hydrogen produced and (2) hydrogen burn phenomenology. The staff expects that thousands of pounds of hydrogen are likely to be generated, while the IPPSS expects hundreds. Further, the staff believes that the loading pressures from a given amount of hydrogen burning, as calculated in the IPPSS, may be low. Both these key areas are discussed further below.

Q.22 Please elaborate on differences in how much hydrogen will be produced.

A.22 The major source of hydrogen during core meltdown accidents is from metal oxidation. For convenience, we consider three stages in the meltdown and discuss differences between IPPSS and staff estimates of hydrogen generation during these stages. The three stages are:

1. After the core is uncovered but prior to core slump

2. When the core slumps into water in the bottom of the reactor vessel

3. During interactions of core debris with water and/or concrete in the reactor cavity.

1. Core Uncovered:

There are virtually no differences between the IPPSS and staff assessments (in NUREG-0850) regarding zirconium oxidation during this phase of the accident. IPPSS concluded that MARCH is conservative with regard to predicting metal oxidation and hence hydrogen generation. However, MARCH was used in the IPPSS during this phase of the accident to predict hydrogen generation, an approach consistent with the staff analysis. (The staff considered oxidation of steel in the core region and found that it did not contribute significantly (<10%) to hydrogen generation during this phase of meltdown. Oxidation of steel structures above the core was considered in the IPPSS (but not by the staff) and also found not to contribute to hydrogen generation when the core was uncovered.) Clearly zirconium oxidation is the dominant source of hydrogen during this stage of core degradation. Consequently, since both the IPPSS and the staff's report use MARCH to predict zirconium oxidation, both studies predict similar hydrogen generation up to the point that the core slumps.

2. Invessel Core Slump

There are major differences between the IPPSS and NUREG-0850 regarding the amount of additional metal oxidation that can occur as the core collapses into water in the bottom of the reactor vessel. In NUREG-0850, we considered that uncertainties associated with the phenomena warranted the assumption of a 100% zirconium/water reaction.

In the IPPSS it is suggested that only an additional 20% of the zirconium would react during core slump for accident sequences with low primary system pressure. For accident sequences with higher primary system pressure, IPPSS assumed an additional 50% of the zirconium would react during core slump. The IPPSS, position was based on a scenario which postulates that silver from the control rods will melt first (silver has a relatively low melting point) and form a plug in the lower, cooler region of the core (silver retains its metallic properties upon melting and refreezing in an oxidizing atmosphere). This silver plug would hold molten core materials as they slump from the central region of the core. The silver plug would eventually fail (locally) and the molten core materials would pass through the lower core support plates without having to sequentially melt them. Water would then be moved out of the bottom of the reactor vessel by the molten core debris. The molten core debris/water interactions would be minimal and any fragmentation would result in formation of relatively large particles. The additional range of metal oxidation assumed is from experimental data based on relatively coarse particles. With minimum core debris/water interaction, the core materials will remain hot (and for the most part, molten) so that local penetration of the reactor vessel will start immediately.

We consider the above scenario to represent just one of a number of scenarios that could be postulated to describe in-vessel core meltdown. The melting of silver and the forming of a plug is an important aspect of the proposed scenario. However, it should be noted that tests at ORNL indicate that silver could be dispersed from the core region as an aerosol. Thus these experiments would suggest a different behavior of the silver than proposed in the IPPSS.

The suggestion that all the molten core will pour through holes in the lower support plates appears to rely on the local melting of a silver plug, which is in doubt. Finally, the size of the particles formed during core debris/water interactions is an area of concern. It is known that small particle sizes lead to faster, more complete metal oxidation. We are therefore concerned at the size of the particles assumed in the IPPSS. Based on recent Sandia Tests, there is a possibility that much finer particles could be formed, which in turn suggests that up to 90% of the metal could be oxidized. We consider that the above discussion adequately illustrates that the scenario proposed in the IPPSS although plausible is simply one possible description (perhaps even a limiting description) of how a core meltdown could progress. There are clearly other plausible scenarios that would involve significantly more metal oxidation. In view of the above considerations, and recognizing our lack of knowledge in this area, we feel that a 100% zirconium/water reaction should be used to determine the hydrogen production during in-vessel core heatup and meltdown.

3. Core Debris/Water/Concrete Interactions

There are again major differences between the IPPSS and NUREG-0850 regarding the amount of metal oxidation that can occur as the core debris is released to the reactor cavity. The IPPSS scenario envisions a high pressure ejection of molten core materials into the reactor cavity which would result in water being driven from the cavity via the instrument tunnel. As the primary system depressurizes, the blowdown forces would disperse the core debris out of the reactor cavity. Minimum core/concrete and core/water interactions would occur, hence minimum metal oxidation and hydrogen generation is proposed. However, the scenario does postulate that 50% of the molten core materials will be brought into thermal equilibrium with the containment building in a very short time. Consequently, the scenario provides a significant pressure pulse in the containment building at vessel failure. Also, the rapid cooling of the core materials requires significant dispersal of the core debris out of the reactor cavity. If the majority of the core debris remained in the cavity, significantly longer quenching times would be predicted with the IPPSS heat transfer model.

The staff considers the above scenario as just one of a number of possible out-of-vessel core meltdown scenarios. The dispersal forces associated with vessel failure are important. It appears important for the core materials to be molten as they exit the reactor vessel. This in turn depends on the mode of in-vessel core slumping and the vessel failure mechanism. The temperature at core slump is an input parameter in the IPPSS scenario. If the core materials were at a lower temperature, significant quantities of the oxides could be solid. In this case, a slurry would be exiting the vessel with quite different fluid properties than the molten materials proposed in the IPPSS study. The lower temperature slurry could also further solidify on contact with the concrete, which would again influence the potential for core dispersal. Even if we accept the dispersal of 50% of the core material from the reactor cavity, the remaining 50% must eventually end up in the cavity. It is not clear how this remaining half of the core (with accompanying steel) can be brought into a coolable debris bed configuration in the reactor cavity without significant additional metal oxidation. There would not be any energetic blowdown forces to disperse or rapidly quench the remaining 50% of the core as it slumps.

We consider that the procedure adopted in NUREG-0850 is appropriate for bounding potential out-of-vessel core meltdown phenomena. We consider the IPPSS dispersal model to be similar to the HOTDROP model discussed in NUREG-0850. Both approaches result in rapid quenching of the core material and virtually no metal oxidation (refer to Figure 3.16 in NUREG-0850). The models therefore maximize the potential for an overpressurization failure of containment at vessel failure. The NUREG-0850 approach posed more of a threat than the IPPSS scenario because 100% of the core materials were assumed to exit the vessel compared with 50% in the IPPSS. However, even with this conservative assumption, NUREG-0850 also concluded that the threat to containment from overpressurization at vessel failure was minimal.

The alternative approach in NUREG-0850 for bounding potential out-of-vessel core meltdown phenomena was the HOTDROP bypass model.

In this model, heat transfer from the core debris to water was assumed limited by critical heat flux considerations and the core debris was allowed to interact with concrete (refer to Figure 3.16 in NUREG-0850). This model assumes that the majority of the core materials remain in the cavity and that several hours are required to quench the core debris. During this time steel oxidation could produce an additional 2000 lb of hydrogen. Pressurization of the containment building is obviously much slower in this alternative approach; consequently, the potential of an overpressurization failure is minimized while the potential for a hydrogen failure is maximized.

We realize that the two approaches suggested above (and in NUREG-0850) represent bounding calculations in terms of maximizing two potential containment failure modes. Calculations in NUREG-0850 indicated out-of-vessel quenching times of less than one hour. It would therefore appear reasonable to suggest pressurization rates and hydrogen generation compatible with these quenching times. This is consistent with the approach taken in Section 3.2.3.4 of NUREG-0850. We therefore believe that 1000 lb of hydrogen should be used as the amount of hydrogen generated during the transition of the molten core materials into a coolable debris bed in the reactor cavity.

Q.23 Could you now elaborate on the differences in hydrogen burning phenomenology?

A.23 In regard to combustion phenomenology, it is impc·tant to mention two aspects of hydrogen problems that appear to be not yet fully resolved. These are:

1. Flame acceleration in hydrogen concentrations of 10 - 12% appear possible in large containment volumes on a fairly extensive scale. Experimental tests at McGill University have indicated that flame velocities greater than 220 m/sec are reached in tubes containing these concentrations with simple obstacles. Pressure may exceed the adiabatic calculations but it is not known whether the time scale of the pressure pulse is sufficiently extended to be a serious containment problem. Experiments on these phenomena are planned at Sandia National Laboratories.

2. For nonuniform hydrogen compositions, although no specific sub-volumes have been identified within the Indian Point containment that would be obviously dangerous from the point of view of collecting explosive mixtures, the circulation and mixing patterns have not been established well enough to preclude their existence.

Thus, although the staff believes that the 3000 lb hydrogen source term adequately represents the principal possible core melt accidents, the full implications of this amount of hydrogen are not yet known because of deficiencies in the understanding of containment combustion phenomena.

Q.24 Please continue your comparison with the IPPSS by noting the differences in the treatment of basemat pentration.

A.24 In the IPPSS, the assessment concludes that a coolable debris bed will be established if the reactor cavity is flooded and supplied with water; thus it is suggested that no extensive basemat penetration will occur under these circumstances. Further the IPPSS assessment disregards basemat penetration for dry-cavity cases, as the containment building is assumed to have already failed by containment building overpressurization (considered a far-worse case). The staff assessment differs from the IPPSS for both the flooded cavity and the dry cavity configuration.

For the flooded cavity, it is the staff position that debris bed coolability is not guaranteed. Thus the staff uses basemat penetration probabilities that range from 10 to 20%. The details of the staff assessment are in NUREG-0850, Section 3.2.3.3.

For the dry cavity case, the staff analysis predicts basemat penetration in about three days. Because this analysis is conservative, that is, that penetration is not a certainty, we have assigned a 10% probability that the basemat will not be breached.

Q.25 Please continue your comparison of the staff analysis with the IPPSS by noting the differences in the key radionuclide release categories. Also compare to the WASH-1400 release categories.

A.25 The release categories A, B, and C, are the potential major contributors to risk. "A" is the release category for seismic containment building collapse event; "B" is the release category for "Event-V and steam-explosions"; and "C" is the release category for the Steam Generator Tube Rupture Event and the slow overpressurization containment building failure model. The analogous release categories from the IPPSS are Z1, 2, and 2RW, respectively. Figures III.B.13, 14, 15 compare the radionuclide-group releases for these three release category groups. For the "B=2" and "C=2RW" release categories, the differences are negligible. For the "A=Z1" set, Figure III.B.13, the NRC releases are all higher than the IPPSS releases. When all the other release categories are taken into account, however, the impact of this difference is not large. In conclusion then, there are no substantive differences between the staff analysis and the IPPSS analysis pertaining to the largest three release categories. It is also instructive to compare the IPPSS and staff release categories to those in the original Reactor Safety Study (WASH-1400) as listed in Table 5-1, page 78, of the main report. The "A=Z1" release category has no equivalent in WASH-1400 since the release category is for an accident that was

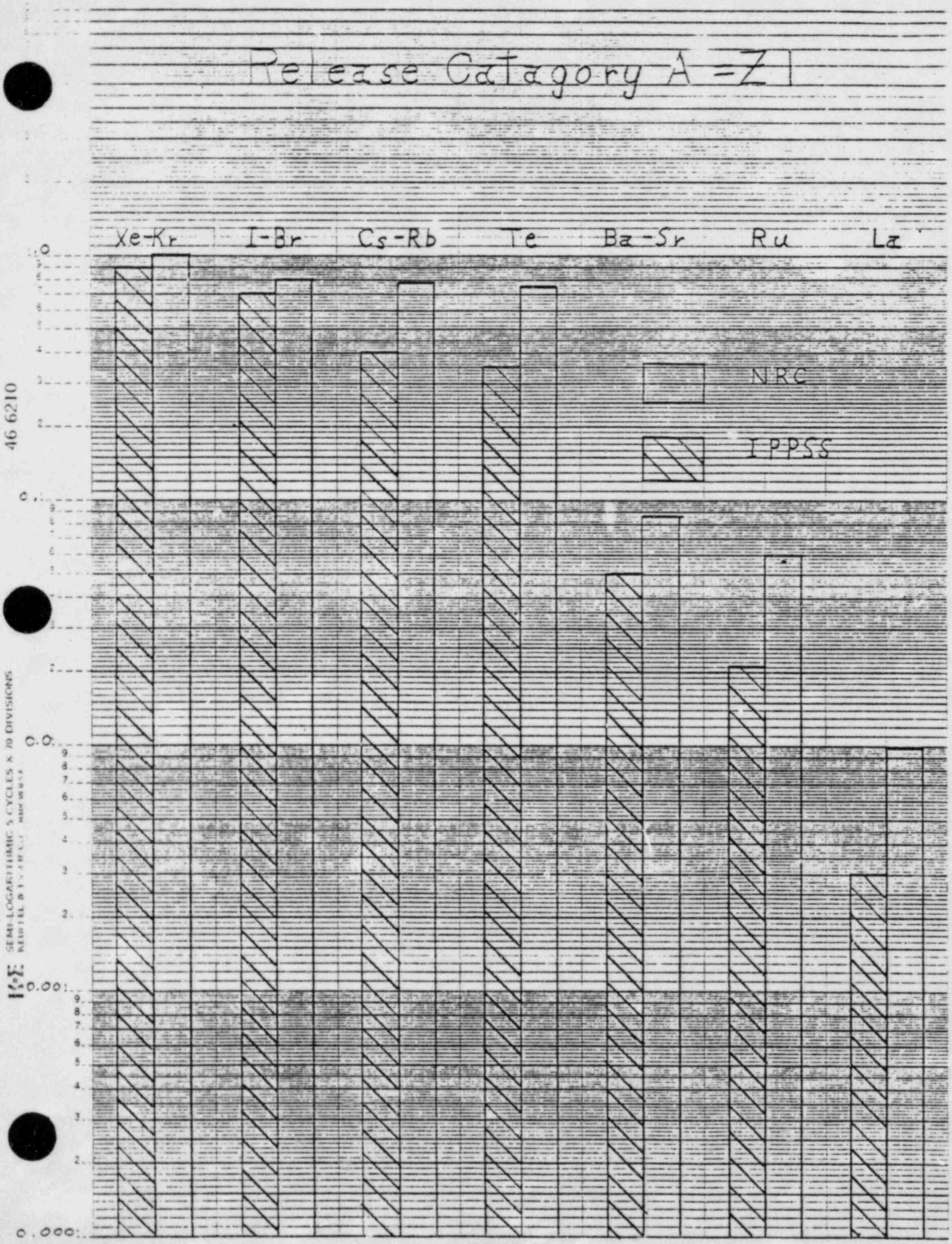


Figure III.B-13

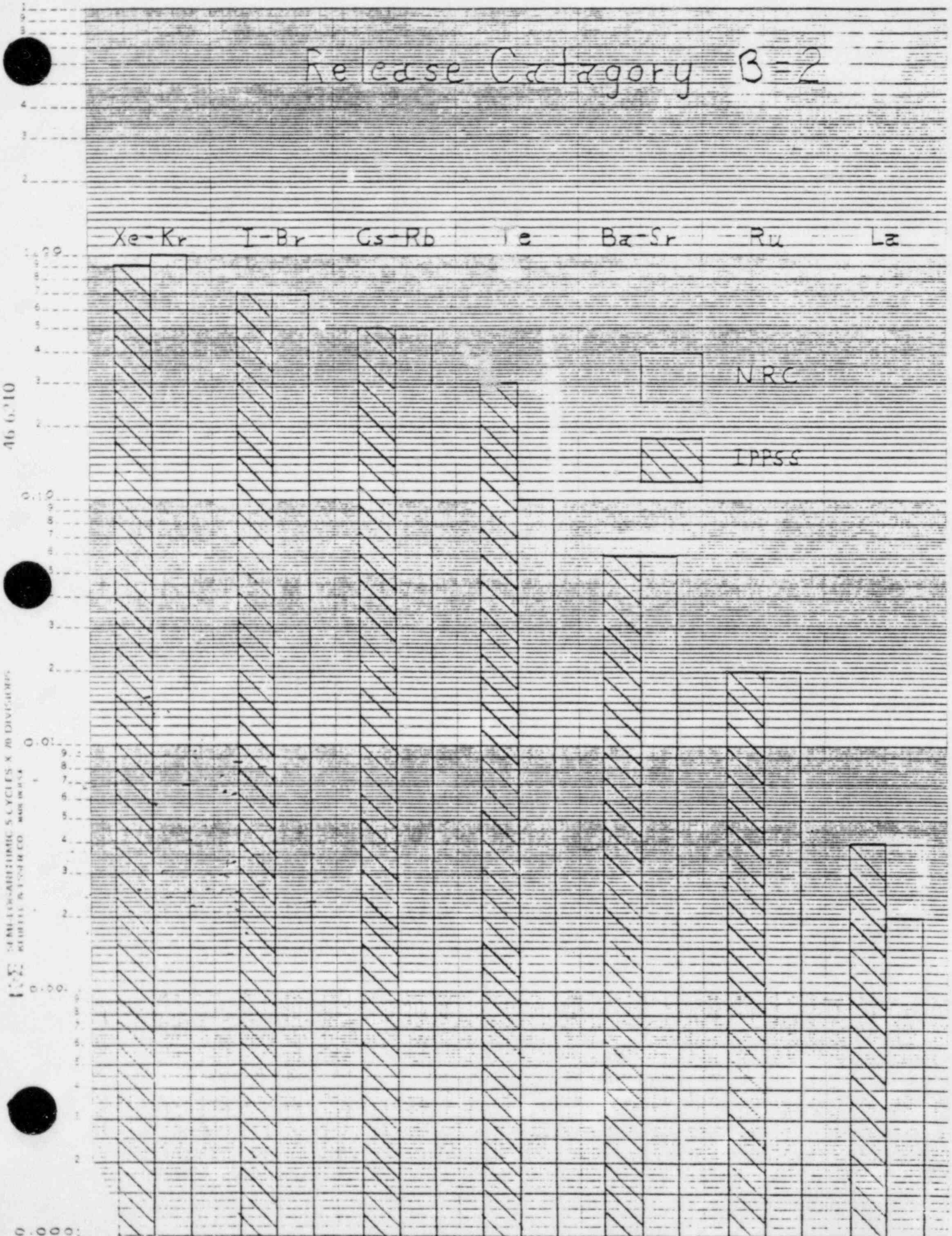


Figure III.B-14

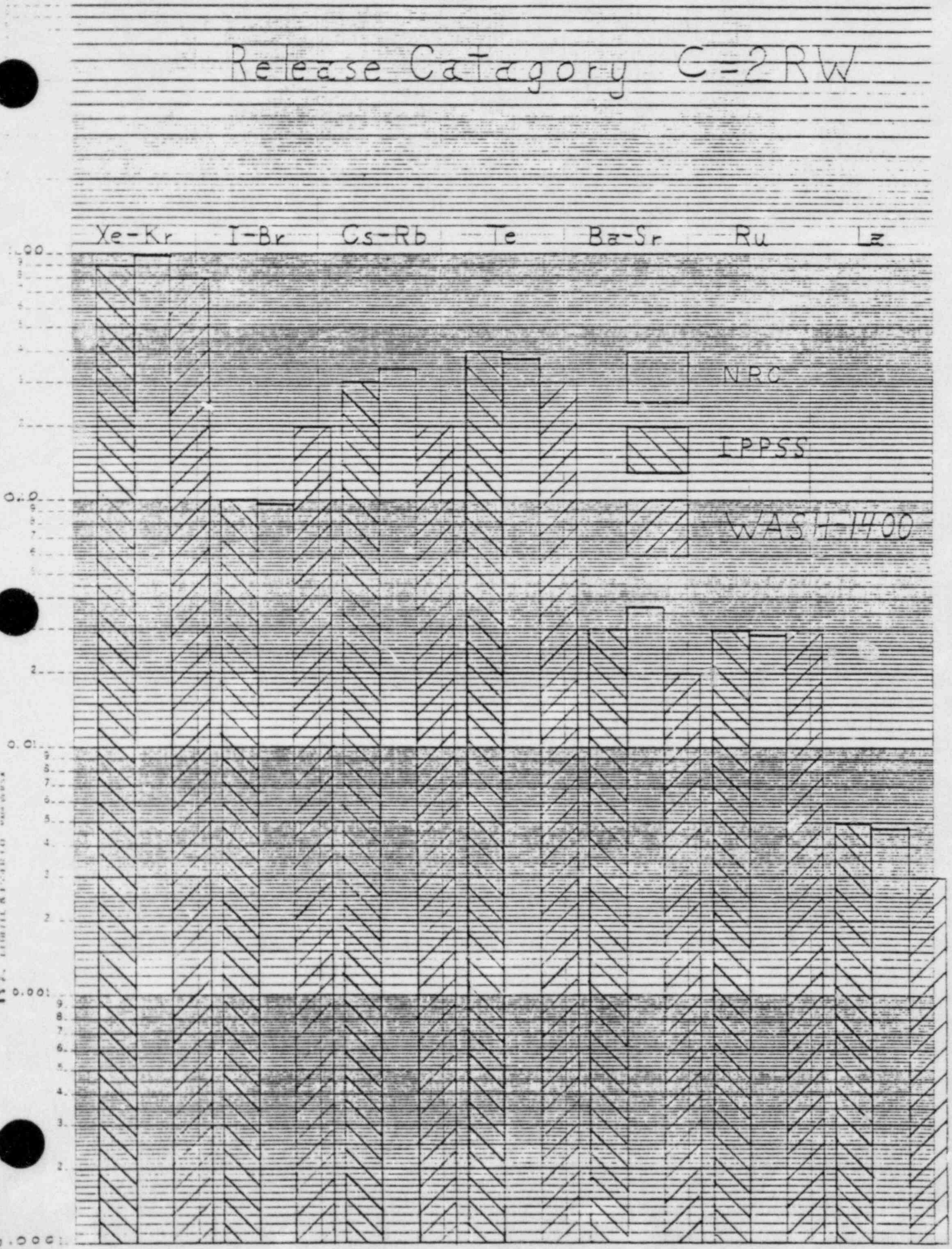


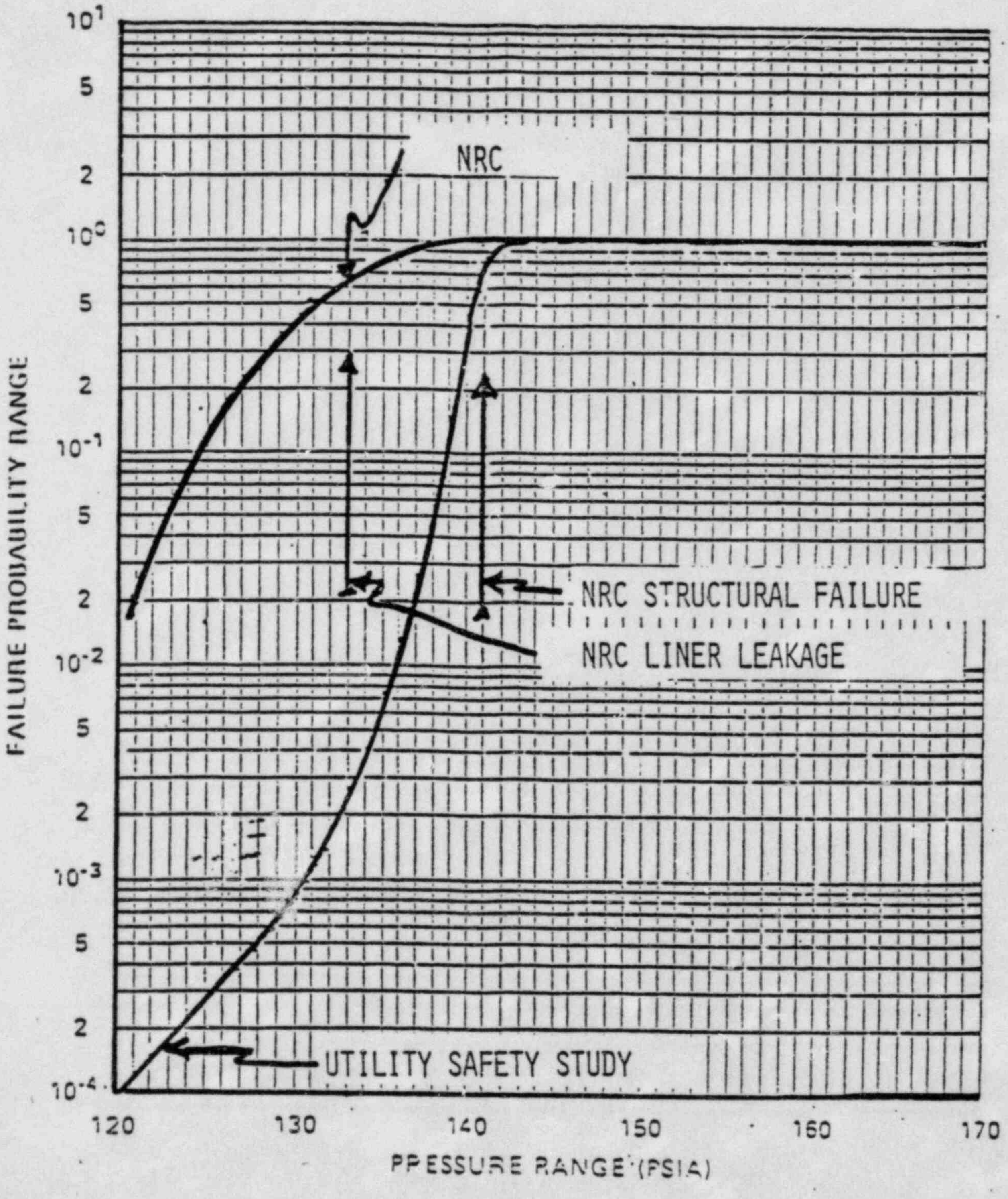
Figure III.B-15

not analyzed in WASH-1400, namely the large seismic event. The IPPSS release category 2 (note Figure III.B.-14) is identical to the WASH-1400 release category 2. This category is reserved exclusively for the Event V accident in the IPPSS study while in the WASH-1400 study it it used for some γ and δ failure modes as well as for Event V. In Figure III.B-15, the IPPSS category 2RW, the staff category C, and the WASH-1400 category 3 are compared. The primary contributor to the WASH-1400 category 3 release is the overpressurization failure; therefore category 3 is the appropriate one to use in this comparison. Except for Iodine, all the WASH-1400 values are approximately equal to or slightly lower than the staff and IPPSS values. If the WASH-1400 release category replaced the staff's release category C in the staff analysis, the net effect would probably be slightly lower risk.

Q.26 Please conclude your comparison by noting the differences in the containment building failure pressures calculated for the two studies.

A.26 In an earlier portion of the Section III.B testimony, we summarized the staff assessment of containment building failure pressure. This failure probability profile is plotted as a function of containment building pressure and compared to that calculated by IPPSS in Figure III.B.16.

From both analyses, the containment building failure pressures are similar for probabilities greater than 80%. However, because of a larger standard deviation in the staff analysis, there is a sizable difference in the two profiles at lower probabilities. This is because IPPSS determined that the only containment failure event of importance is extensive yielding of key structural elements (in this case, the rebar) while the staff considered failure of the liner and leakage through penetrations at lower pressures in addition to the yielding of rebar considered by IPPSS. It is difficult to determine the impact of using the IPPSS failure values in place of the staff's values. However, it is clear that the slow overpressurization events would occur somewhat later and that fewer hydrogen burns would result in containment building failure.



INDIAN POINT CONTAINMENT FAILURE PRESSURE PROBABILITY DISTRIBUTION

Figure III.B-16

APPENDIX A

Discussion of Containment Event Tree Split Fractions

A) LF Damage State at Branch Point 2 for Hydrogen Burns: Potential for hydrogen-burn containment building failure prior to vessel melt-through:

For the LF damage state, the partial pressure of steam in the containment building is low. Also, core-heat-up is slow relative to other accident damage states. The maximum hydrogen release to containment prior to vessel failure would be limited to ~2000 lb (~100% metal-water reaction). There is a good potential for this hydrogen to burn at the lower flammability limits; however, if we assume that all the hydrogen burns adiabatically, a pressure rise of only 65 psi would be produced. If this is added to the pressure in containment (of ~20 psia before the burn), the final pressure after the burn is only 85 psia, which is significantly below the failure pressure of the containment building. We are, however, concerned about the possibility that local pockets of hydrogen could form and raise the potential for damaging detonations to occur. Containment could be threatened indirectly through high temperature damage to safety systems. Although the adiabatic burn does not fail containment, the uncertainties associated with combustion of 2000 lbs. of hydrogen warrant a split fraction of 0.01 at this branch point.

B) EFC and EF Damage States at Branch Point 2: Potential for hydrogen burn containment building failure prior to vessel failure

The potential exists for a hydrogen burn prior to vessel failure for these sequences as well as for the LF damage state. However, core heat-up and slumping is much faster for the EFC and EF damage states. The primary system pressures are also higher so that less hydrogen is released prior to vessel failure. Also, the vessel failure time is shorter for the EFC and EF damage states. This implies a lower probability of a large hydrogen burn prior to vessel failure for damage states relative to the LF damage state. Thus we assume that the hydrogen burn effect is negligible and assign zero to the split fraction.

C) EFC and EF Damage States at Branch Point 8: Hydrogen-burn containment building failure after vessel failure

If we assume that hydrogen did not burn prior to vessel failure, it is possible that 3000 lbs. of hydrogen could be available shortly after vessel failure. The steam spike associated with vessel failure amounts to about 40 psia. The mole fraction of steam is close to the value that would render the containment atmosphere inert. At this pressure, the containment spray system would be actuated for the EFC damage state, bringing down the steam partial pressure and rendering the containment building combustible. If we assume 3000 lbs. of hydrogen burns adiabatically, a pressure rise of 95 psi would be expected. However it is known that a number of mechanisms will tend to limit the actual pressure rise associated with hydrogen burning to less than the theoretical adiabatic limit. A computer code (HECTR) has been developed at SNL to calculate the actual pressure rises associated with hydrogen burning by considering, heat transfer by radiation, convection, and spray-droplet evaporation. Calculations with HECTR indicate that only 80 to 90% of the adiabatic pressure rise will actually occur. Based on these calculations and noting that the containment sprays will be operating for the EFC damage state, the split fraction of 0.03 for a positive response at branch point 8 was selected. However, the above assessment has recognized uncertainties. Considering that the burn may not occur at the upper pressure limit of 40 psia and recognizing that more or less than 3000 lb of hydrogen may be produced, we consider the impact of a more optimistic assessment and more pessimistic assessment in the assessment of uncertainties as described in this testimony.

For the EF damage state the conditions in the containment building would be similar to the EFC damage state up to the point of vessel failure. However the sprays are assumed not to operate for the EF damage state (unlike the EFC) so that there would be no water droplets from the sprays to contribute to reducing the pressure rise associated with a hydrogen burn. Consequently, for the EF damage state we assign a split-fraction of 0.10 for a positive response at branch point 8 compared with 0.03 for the EFC damage state. The higher split fraction for damage state EF recognizes the lack of spray operation and also that the burn, even without sprays, will not be adiabatic.. We also consider the uncertainties in this value as we did for the EFC damage state.

APPENDIX B

Unreliability and Negative Characteristics of Mitigation Features

I. Hydrogen Control Using Glow Plugs

The unreliability of the glow-plug system used in this study is 5%. The value appropriate for similar systems installed in ice-condenser type containment buildings (e.g., Sequoyah) is lower (less than 1%). However, we choose the more conservative value of 5% here to account for performance uncertainties resulting from the more hostile environment of a core-melt accident. This unreliability factor is used in the containment event tree in the following way. For the relevant containment event tree branches (2, 8), the probability of containment failure "before mitigation" is multiplied by 0.05. That is, if the original building failure probability was 10%, the new probability for failure is 0.5%. (Note that this probability is 0.0% for ideal glow plugs, i.e., those that perform flawlessly.)

II. Passive Containment Heat Removal Using Heat Pipes

The unreliability of a heat pipe system used in this study is calculated at 5%. Ordinarily, unreliabilities for containment heat removal systems are lower; however, because this is a new system with no record of performance capability under the accident conditions, a larger unreliability value was assigned. Crud buildup on heat transfer surfaces was determined to be the major contributor to degraded performance or unreliability. This unreliability value is then used to reevaluate the failure probability at branch point 9.

III. Basemat Penetration Prevention Using a Flooded Cavity

The unreliability of this core retention system is calculated at 5%. This value is probably too low; that is, the probability of not achieving a coolable debris bed and thereby not preventing basemat penetration is probably higher based on phenomenological considerations. The impact of large variations in this unreliability parameter is not great since the impact of basemat penetration on overall risk is small anyway. The major impact in considering core retention results from the negative features associated with flooding the reactor cavity. Flooding the cavity is essential for core retention: however, the following negative features must be taken into account.

- by flooding the reactor cavity the probability of failure by overpressurization increases
- by flooding the reactor cavity the potential for release of contaminated sump water through the basemat increases.

The first negative characteristic is so important that such a core retention system could never be part of the containment design unless a containment heat removal system was also included. Without providing for containment heat removal, this core-retention system would actually increase risk because of the increased potential for overpressurization failure.

The second negative characteristic would only be important if the liquid pathway for the Indian Point site was an important route for distribution of radionuclides to the environment. In the testimony of Richard Codell, it is determined that the liquid pathway is not an important risk consideration.

Q. Dr. Meyer, would you please provide a brief summary of your testimony?

JUDGE GLEASON: I think, unless there is something, we have been waiving this part of the hearing, Ms. Moore. Unless you want to put something on the record specifically, I would just as soon as go directly to cross examination.

MS. MOORE: The witness is available for cross examination.

JUDGE GLEASON: Mr. Blum?

MR. BLUM: Thank you, Your Honor.

CROSS EXAMINATION BY MR. BLUM:

Q. Dr. Meyer and Dr. Pratt, which version of CORRAL was used for your testimony?

A. (Witness Pratt) CORRAL 2.

Q. On page 3 B 24 of your testimony at the very top paragraph you state, "In order to make further calculations manageable it was determined that the nine release categories listed would adequately represent the large number of calculated releases." How was that determined?

A. (Witness Meyer) We had more than nine release categories. However, past experience has indicated that the nine that we selected were sufficiently representative of the various types of containment failure modes to propet.y reflect the release category characterization for the analysis.

JUDGE SHON: Dr. Meyer, I am not sure that answered Mr. Blum's question. You assured us the nine odes you used were representative of all the many different combinations, but how did you assure yourself that they were, indeed, representative, that there were not half a dozen that you hadn't accounted for that were very very different?

THE WITNESS: (Witness Meyer) The release categories are listed on page 25 of the testimony. Release category A, the most severe release category, represents a very severe release category.

I don't know of any release category to date that is more severe than that, and that would certainly cover the top end of the key risk category spectrum.

The release category I, on the other hand, was a release category calculated for the no failure case, where we assumed an one percent per day leakage rate from the containment.

For release categories B through H

the release fractions are constantly reduced going from B to H.

Any release category in addition to these nine would fall somewhere in the range from B to H, and we are close enough to the release categories indicated to have but a very small impact on the direct consequence analyses.

JUDGE SHON: What you seem to be telling me, see if I have this right, is that you selected release categories that had a broad range from the worst to the least significant of consequences, and that the ones that you selected in between, which would have had intermediate consequences, were in some sense, and I am not sure what that sense is, not far from the others. Is that it?

THE WITNESS: (Witness Meyer) That's correct.

JUDGE SHON: It's a little complicated in one's mind to decide whether there might be a category that looks for all the world like D, or something like that, but has the sufficiently different makeup to it that it would -- that it would have some effect on your calculations or your estimates. Do you see what I mean?

THE WITNESS: (Witness Meyer) Yes. I see your point. And we have performed analyses to convince ourselves that there would be no major variation in our results due to that type of consideration.

In fact, with some hindsight we could have collapsed these categories further. There is a limitation in the contract code that suggests keeping the release categories below ten is desirable from a calculation standpoint.

JUDGE SHON: Thank you.

I think it satisfied me, Mr. Blum. I don't know about you.

MR. BLUM: Yes. That's quite good.

Q. On page B 54 of your testimony, about two thirds of the way down in the first paragraph, you state, "We consider that the above discussion adequately illustrates that the scenario proposed in the IPPSS although plausible is simply one possible description (perhaps even a limiting description) of how core meltdown could progress."

What do you mean by the phrase "perhaps even a limiting description"?

A. (Witness Pratt) This was a

description specifically related to the amount of hydrogen that could be produced during a core-melt event, and we felt that the scenario discussions was a limited valuation in the direction of the smallest amount of hydrogen that it could produce.

We think that there is a possibility for producing more hydrogen. So limiting in the sense of a small amount of hydrogen.

Q. So as far as consequences of an accident are concerned the IPPSS treatment of this phenomenon would be on the optimistic side?

A. (Witness Pratt) Yes.

Q. Under the gamma failure mode the testimony notes that hydrogen burns can induce containment failure indirectly by causing the failure of any safety features geared to protect the building containment function. You recall that, do you not?

A. (Witness Meyer) Yes, I do.

Q. And this suggests that hydrogen burns might cause leakage failures or failures of containment cooling systems, does it not?

A. (Witness Meyer) That's correct, yes.

Q. What review have you performed, if any, of the impact of hydrogen burns on engineering safety feature equipment?

A. (Witness Meyer) We performed a review of the impact of hydrogen burns on the fan cooler system and the spray system, and it was our determination that the systems could survive the burns that we considered.

However, I should point out that we did a parametric analysis in the testimony that explored the impact of engineered safety features failing, and the end result of that parametric analysis was that the impact on risk isn't all that severe

JUDGE SHON: Before we get off this particular point, Dr. Meyer, did any of the burns that you included in your analysis result in formation of a shock wave, or anything like that?

THE WITNESS: (Witness Meyer) The molt fraction of hydrogen necessary to go into the dynamic range, the shock development range, is 18 to 20 percent.

Our analyses indicated that the molt fraction ranges would be in the range of 4 to 16 percent, so we do not think that that type of dynamic leading is a plausible event for Indian Point

JUDGE SHON: Even if the hydrogen were somehow concentrated in some small area or region?

THE WITNESS: (Witness Pratt) In fact, I was just going to bring that point up. One would have to go to local concentration to get that situation.

We did put into the CORRAL indication a residual probability that that would secure, but it was a relatively small value. It wasn't that large. That's simply because the damage states where this would occur are relatively low in probability.

So we are talking about situations where we have containment heat removal systems operating, and in terms of how they differ from the probability it's relatively low.

JUDGE SHON: But once you considered the probabilities of that sort of thing were low, you didn't- --

THE WITNESS: (Witness Pratt) Yes, that's true. We have done quite a bit of work at Brookhaven to get from defraction to detonation. I have a number of publications out about that, but we haven't specifically looked at that at

Brookhaven, at least not in my group.

Q. Included in your review of effects on ESF equipment did you specifically consider electrical cables?

A. (Witness Pratt) We have a number of publications out in that area from Brookhaven, where one of our consultants from Stonybrook looked at that effect, and came to the conclusion that those cables that were exposed could be ignited under a hydrogen burn situation.

Q. I am sorry, I misheard a word?

A. (Witness Pratt) Ignited.

Q. No. You said that they could be?

A. (Witness Pratt) Could be. That's right.

Q. And did you pursue the possible effects of their being ignited?

A. (Witness Pratt) The possible effects, in terms of the survivability of equipment, we don't think that particular equipment would have been relying on those cables at this stage of the accident.

JUDGE SHON: Are you saying, Dr. Pratt, that the cables would burn, but there would be no safety equipment relying on them?

THE WITNESS: (Witness Pratt) For these severe accidents, I think that's the case.

Well, if there's a confusion, I think we are relying on, say, the operations of the containment system to function and cool, to maintain the building's integrity. Whether or not we have the circuitry necessary for ECC. That's the point I am making

JUDGE SHON: But I am not sure how you identify, as it seems you would have to, that each and every cable which was exposed and could burn was a cable that was important only to an earlier stage in the history of the development of the accident, or at least was not necessary as a mitigating feature from that point on.

Wouldn't you have to identify these things, each and every one, and say yes, and no?

THE WITNESS: (Witness Pratt) I would agree. The calculations we made at Brookhaven was to look at the temperature environment associated with a hydrogen burn, to look at the materials that were made up of the cable material, and see whether it would ignite.

And we came to the conclusion that certain materials would ignite under those circumstances.

JUDGE SHON: But that answer, would you not have to analyze the specific situation to decide that each and every cable which could so ignite was a cable not important to the history from that point on of the accident?

THE WITNESS: (Witness Pratt) Yes, you would.

What I am saying is that in our testimony we explored in a parametric factor those systems failing, and we found it was not a large factor.

JUDGE SHON: Then you did do that.

THE WITNESS: (Witness Pratt) Well, I can't say I went through and checked he have -- we simply said okay, let's assume everything was lost, and did the analysis using that assumption. This is discussed on page 49, that the effect was not large.

So it was not a detailed analysis on my part or any people in my group as to which cables were important. We simply assumed these systems were lost for this damage state.

JUDGE SHON: You assumed they were all lost?

THE WITNESS: (Witness Pratt) Yes, sir.

JUDGE SHON: I see. Thank you.

Q. Are not the cables to the containment spray and the fan coolers inside containment?

A. (Witness Pratt) My answer was I don't know.

A. (Witness Meyer) As I understand, there are cables for the fan cooler systems inside containment.

Q. Do you recall at all for the sprays?

A. (Witness Meyer) No, I do not.

Q. You stated that there were certain accident sequences where you assumed the safety equipment ESS would be inoperative. Is that correct?

A. (Witness Meyer) Yes, that's correct.

Q. And then there were other sequences where you assumed that there was no probability of them being inoperative. Is that correct?

A. (Witness Meyer) That's correct, yes.

Q. Were there any sequences where you engaged in any sort of probablistic modeling that they would become inoperative during that sequence?

A. (Witness Meyer) We would have done

that if there was a demonstration of significant sensitivity to that particular consideration.

Since our parametric analysis that we felt scoped the problem, the indication was that there was not that sensitivity, we didn't pursue the matter further.

Q. Which of the sequences where you found the sensitivity?

A. (Witness Meyer) On page 49 of the testimony, you will note that we performed an analysis assuming that the cooling fans failed in the LF damage state 25 percent of the time.

If they could fail 25 percent of the time, then the results shown would come about, namely they would have negligible effects on early fatalities, and we would get anywhere from a 30 percent to a 60 percent increase in risks if you are using the late fatalities risk more than you are.

Q. Are all instances of hydrogen burns contained within the LF damage state?

A. (Witness Meyer) No. They can be also contained in any of the damage states that have containment cooling, including the EF and the EFC damage states.

Q. So there were some instances of hydrogen burn where you assumed a zero probability of the ESF equipment failing. Is that correct?

A. (Witness Meyer) You mean in the context of this particular parametric study?

Q. Yes.

A. The footnote indicates that a similar analysis was performed for the EF damage state for both units, and both risk measures the changes in risk were negligible.

Q. But I believe you also mentioned the EFC damage state?

A. (Witness Meyer) Yes. For that particular damage state you would have to assume that you lost both your fans and your sprays, and we felt that there was negligible chance that that would occur.

In addition, in another area of that uncertainty analysis, we increased the probability of hydrogen failure by an order of magnitude, and even for that rather gross parametric assessment in terms of conservatism, the indications are that the risk value is changed by no more than a factor of two.

Q. Under the beta failure mode you quote a ten to the negative three conditional probability of conditional failure to isolate probability. What is the source of this number?

A. (Witness Meyer) This number was provided me by the reliability and risk analysis branch within NRR.

Q. Is this the generic number or is it Indian Point specific?

A. (Witness Meyer) As far as I understand, it's an Indian Point specific number.

Q. Do you know how it was derived?

A. (Witness Meyer) No, I do not.

Q. Did you explore parametrically the impact of higher probabilities?

A. (Witness Meyer) We did not do a formal parametric analysis assuming higher beta failure mode probabilities.

However, it would be my judgment that that failure mode would have to increase considerably for it to start having an impact on the overall risk at Indian Point.

A. (Witness Pratt) If I could add to that, although Jim said we did not submit a formal analysis, I have done the calculations, and even if you increased that to a probability of one, and made it equal to the core-melt probability, it would not be a main impact.

Q. Under category beta prime the testimony discusses seismic failure of the containment.

Did you evaluate the seismic capability of hatches, such as for personnel and equipment?

A. (Witness Meyer) For that particular failure mode, the beta star, we assumed that the containment fails at the initiation of the accident, so the question of integrity of the hatches is irrelevant.

Q. You are aware that the licensees claim that there can be no direct seismic failure to the containment?

A. (Witness Meyer) I understand they recently submitted an amendment to the IPPSS that claims that, yes.

Q. Have you evaluated that?

A. (Witness Meyer) No, I have not.

A. (Witness Pratt) I haven't, either.

Q. Why haven't you evaluated it?

A. (Witness Meyer) Well, first of all, it's not in my area of responsibility to review the seismic portions of the analysis.

Second, it came in too late to have it formally incorporated into this proceeding, at least in terms of my containment analysis.

Q. As far as either one of you knows, there has been no formal staff evaluation of this matter?

A. (Witness Meyer) That is -- I just don't know.

Q. But insofar as there are portions of IPPSS amendment one that would relate to your area of expertise, you have been unable to evaluate it because of the late date at which it came in. Is that correct?

A. (Witness Meyer) If I understood --

JUDGE GLEASON: Excuse me. Could I hear that question again, Mr. Blum?

Q. Insofar as there are portions of IPPSS amendment one that relate to your expertise, the reason you have not been able to evaluate it is the late date at which it came in. Is that correct?

A. (Witness Meyer) Could you clarify what you mean by that material related to my expertise?

Q. Well, it would seem that some of the claims in IPPSS amendment one would be in some ways dependent on proper positions about containment integrity and operation of different failure modes in the plant. That's true, is it not?

A. (Witness Meyer) Yes, that's true.

Q. And in some sense both have you would be rather experts for that material?

A. (Witness Meyer) If we were provided with new damage states based on consideration of that amendment, then yes, we could proceed with doing the appropriate containment analysis associated with those changes.

Q. Do you know why you have not been provided with new damage states?

A. (Witness Meyer) Our results, in particular for the beta star failure mode, would be conservative in the sense that if this failure mode is removed the overall risk would be reduced. So in the sense of covering the assessment from a standpoint of conservatism, that analysis has already been provided.

Q. Well, I am asking something slightly different, which has to do with a staff evaluation of IPPSS amendment one.

Do you know why you have not been provided with the damage states that you would need in order to do a critical review of IPPSS amendment one?

MS. MOORE: Mr. Chairman, I object. This is beyond the scope of Dr. Meyer's testimony, he has already said it is not in his area of responsibility to review this statement and make decisions.

JUDGE GLEASON: Well, I think it's a relevant area of inquiry. I think I would like to hear a response.

A. (Witness Meyer) Will you repeat the question, please?

Q. Do you know why you have not been provided -- Earlier you just stated that if you had been provided with new damage states based on IPPSS amendment one you could then utilize your expertise to evaluate some of the processes by which the conclusions were drawn. Is that a fair characterization of what you said?

A. (Witness Meyer) Yes.

Q. I was now asking if you knew why you had not been provided with new damage states based on IPPSS amendment one?

A. (Witness Meyer) I do not know all of the reasons.

As I said before, what we did would turn out to be a conservative analysis if that particular damage state is removed, based on the amendment submitted by the utilities.

Q. Do you know any of the reasons why you were not provided with new damage states?

A. (Witness Meyer) The main reason, I think, is that there was just no time, and it was felt to be not a terribly important issue in regard to the overall question of risk at Indian Point.

JUDGE GLEASON: I believe you are speculating here, Dr. Meyer. Is that correct? I think your testimony indicated that you didn't know why, but that you felt that if there was a reason you felt that it wasn't necessary because your analysis had already been made. Now, is it your testimony that you do know why?

THE WITNESS: (Witness Meyer) No. I would leave it at that.

JUDGE GLEASON: I think he has already responded to the question.

Q. You are aware, are you not, that the licensees are claiming rather major reductions in overall risk based on IPPSS amendment one, are you not?

A. (Witness Pratt) Yes.

JUDGE SHON: Mr. Blum, I guess I am still a little confused. As I understood Dr. Meyer's testimony he said that IPPSS amendment one claims a reduction in risks. But his testimony as far as the testimony before us today simply doesn't allow for that reduction in risk. It leaves the risk just the way it was before IPPSS amendment one. Isn't this true?

THE WITNESS: (Witness Meyer) I assume you are referring specifically now to the beta star failure mode, and I made my comments in that context.

JUDGE SHON: That's the only thing that has come up. The beta star failure mode is, in effect, removed in IPPSS amendment one, and it is not necessary for you to know what the none damage state is when you don't damage it that way, simply because you have said it's damaged. Is that correct?

THE WITNESS: (Witness Meyer)

That's basically correct.

Q. Were there any other respects in which you were able to evaluate the significance of IPPSS amendment one, apart from beta star?

A. (Witness Meyer) Again based on what I understand the amendment to be, the answer is no.

Q. Do you know whether there are any other staff witnesses who will have been in a position to evaluate the significance of IPPSS amendment one?

A. No, I am not aware of any staff witnesses.

Q. Is it the belief of both of you that there are none?

MS. MOORE: Could we have clarification of that question? None what?

Q. Is it the belief of both of you that there are no staff witnesses in this proceeding who will have been able to evaluate IPPSS amendment one?

A. (Witness Meyer) Other than the information that had been presented before this board under question one, no, as far as I know, there are no other witnesses to speak to that amendment.

Q. Is there specific testimony that you are referring to under question one which does specifically evaluate portions of IPPSS amendment one?

A. (Witness Meyer) I can't answer that question until I can get a clarification of the total content of amendment one. I have only been talking about the beta star portions, and I am not familiar in detail with the other components of that amendment.

JUDGE SHON: Ms. Moore, have we had before us staff witnesses who addressed themselves to the remainder of IPPSS amendment one? It doesn't raise any image in my mind, and I frankly don't know.

MS. MOORE: There are no witnesses who have specifically amendment one. The only context, and I, myself, am not familiar with everything in amendment one, but the only context where we would considered certification is, for instance, the bumper between the unit 2 control building and unit 1 superheater building.

JUDGE SHON: Well, that's certainly made very clear by your witnesses. I simply didn't remember whether or not anyone had addressed the amendment as a whole.

What is the date of the amendment, does anyone know?

MS. MOORE: I believe it's somewhere early in February.

JUDGE SHON: Thank you. That puts a good time frame around it for me.

It seems, Mr. Blum, that the staff hasn't addressed it at all.

Do you intend covering that in any way?

MS. MOORE: No, sir, I don't believe we do.

JUDGE SHON: Thank you.

MR. BRANDENBURG: Judge Shon I am reluctant to interfere, and the last thing I want to do is testify.

I think there might be some confusion between addressing amendment one, qua document, and the uotimate question that we are concerned with now. The underlying changees to the plan, themselves, were made in the latter part of 1982, and documents relating to them were provided to the Sandia witnesses that appeared before the Board and various staff witnesses prior to the

Albuquerque meeting that was held, if memory serves me, in October of 1982.

So again I am really not seeking to testify, I am seeking to clarify what strikes me as an ambiguity that the changes to the plans that were imbedded in the licensees' question 1 testimony, and also addressed in the staff's testimony were made known to and discussed with the staff while the amendment one that formally modified the IPPSS study did not mature into a final product, qua a document, until later.

JUDGE SHON: In other words, what you are saying, and I appreciate your clarification, is that although IPPSS amendment one has not been addressed by any set of witnesses as a document, per se, nevertheless, the pertinent changes incorporated into that document, which have been incorporated in the plan, were addressed by the Sandia witnesses and the staff witnesses?

MR. BRANDENBURG: Yes. And made known to them long prior to their testimony.

MS. MOORE: Your Honor, there should be one clarification. Except for the release category A part of amendment one, which is the beta star failure mode, which we have not addressed --

JUDGE SHON: With the caveat, then, that the release category A, beta star, the thing that we have been talking about, that has not been discussed in previous testimony before us. Is that correct?

MS. MOORE: That's correct.

JUDGE GLEASON: We are getting a lot of testimony on this record from attorneys, and I just don't think it's desirable or fair.

I recall a prior discussion about this amendment in testimony with respect to it, and issues and claims of unfairness at that time presented by some of the representatives, and I just don't have the transcript in front of me, of course, to recall it. And I think that there were questions on the part of the Board asking for any amendments to any part of the IPPSS study to be brought forward.

I think that if you have witnesses coming up, if you want to respond to these things, do it through witnesses, and not through your own testimony, please.

Mr. Blum, back to you.

Q. Beyond assuming that the containment

fails due to seismic activity, did you do any analysis of particular mechanisms by which the containment could fail?

A. (Witness Meyer) Yes, we did. The containment will receive a certain pressure history, pressure loading, as well as temperature history, temperature loading. And based on these considerations we explored the various containment failure modes.

We have spent sometime in past testimony, cross examination, describing the structural containment analysis that led us to the conclusion that 126 p.s.i.g. is a good number to use for our estimate of the failure of the pressure of the containment.

Q. Did you specifically consider possibilities of some sort of intermediate failure between gross structural failure of the rebar and nothing occurring, no failure?

A. (Witness Meyer) Yes, we did. In fact, we assumed that below this 126 p.s.i.g. value, namely at 116 p.s.i.g., that the containment would fail 50 percent of the time. We did this in order to take into account the possibility that the containment would fail in terms of extensive leakage before it saw the 126 p.s.i.g. pressure.

Q. Did you examine how the containment fails under seismic loading?

A. (Witness Meyer) The only time we considered the seismic failure of containment, that is direct failure of the containment due to the seismic event, was under the beta star release. That is release category A.

The assumptions we made there was that there was such an extensive gross leak or failure of the containment that the radio nucleide would be released very rapidly

Q. Did you specifically examine the mechanism by which the containment fails? For example, is it the rebar, is it the hatch, and so forth?

A. (Witness Meyer) The 126 p.s.i.g. value is the value that we used for -- that we have determined to be the point at which there is extensive yielding of the rebar.

We considered also various other mechanisms for failure. We have already discussed at some length the leakage failure that would ensue from a failure of the liner of the containment liner at lower pressures.

There are, of course, penetrations, and we have considered the possibility of leakage to these penetrations at lower failures.

Q. I am sorry. I am referring specifically to the seismic event, where it's the ground acceleration acting on the structure, and what I would like to know is whether you performed any analysis of the mechanism by which containment fails under those circumstances?

A. (Witness Meyer) I did not perform any analysis of that.

Q. Does that answer stand for both of you?

A. (Witness Pratt) Definitely.

Q. With regard to page B 22 of your testimony, you stated that you assigned a split fraction of zero to the event of regaining ac power for damage state E.

Isn't it true that in other places of your testimony you at least imply that there is a small probability of regaining ac power in time to effect outcome?

A. (Witness Meyer) You would have to refer to that specific portion of my testimony.

Q. Well, at this point would you agree

that there is some small probability of regaining ac power?

A. (Witness Meyer) Yes, there is some small probability of regaining ac power.

Q. All right.

The specific portion of the testimony, just to clarify, is on page B 22, the second to the last sentence of the first paragraph, where it says, "Either power would be restored prior to correspond uncovery, thereby preventing correspond degradation, or, if not then the probability of power restoration-occurring early enough to effect the outcome of the E damage states is small."

Do you see that?

A. (Witness Meyer) Yes. That's a correct quote from the testimony.

Q. Is not assigning a split fraction of zero here a rather optimistic assumption under the particular factor of damage state E?

A. (Witness Meyer) Well, it would have two cancelling effects. If you establish containment cooling, then you decrease the possibility of pressurization failure resulting from steam and noncondensible. Otherwise, restoration of ac power can bring you from an inerted containment condition to a deinerted containment condition, and thereby the possibility of hydrogen burns.

Q. And isn't it in some ways optimistic to assign a probability of zero to that event occurring?

A. (Witness Meyer) This optimistic value was provided to me by the people that determined the probability of restoration of ac power, and it was based on their judgment that we did not include it formally in our analysis.

Q. Who were the people who provided it to you?

A. (Witness Meyer) The staff and the reliability and risk assessment branch.

Q. Thank you.

In your testimony generally a rather large number of pages are devoted to considering the risk reduction effects of the staff mitigation package, were they not?

A. (Witness Meyer) That's correct, yes.

Q. And just for clarification, would you state what that package consists of?

A. (Witness Meyer) To explore the question of risk reduction we considered a passive containment heat removal system, specifically a heat pipe heat removal system, together with glowplugs to control the hydrogen burning, and with the additional requirement of a flooded reactor cavity to give further assurance that the basemat would not be penetrated. Those are the three.

Q. Why did you undertake such detailed examination of these three and their risk reduction potential?

A. (Witness Meyer) Because it was our charter to do so, to explore various candidates and combinations of candidates to provide that key element in the decision making process, namely the risk reduction afforded by such a strategy, or by individual features.

Q. When you refer to your charter, could you be more specific about which instructions these are, and from whom?

A. (Witness Meyer) It evolved from the designed Indian Point study that we have discussed previously, starting in December, 1979.

The charter was laid out in a task action plan that was printed in the winter of 1980, a copy of which has been provided to all parties.

Q. In your professional judgment are there any features of the Indian Point plants or the Indian Point site that would warrant careful exploration of mitigation measures for these plants specifically, apart from having careful examination on a generic basis.

JUDGE GLEASON: Mr. Blum, would you please repeat that question?

MR. BLUM: Maybe I should try to simplify it, too.

Q. In your professional judgment are there aspects of the Indian Point plants or the Indian Point site that would warrant detailed consideration of mitigation features for these plants specifically, even apart from such consideration on a nationwide generic basis? ?

A. (Witness Meyer) I think our conclusions, the staff conclusions, under question 5 are quite clear, that it is our position now to discontinue pursuit of mitigation specific to Indian Point in the context of this proceeding, and to fold the question of mitigation features into a more generic study of mitigation features for reactors in general.

Q. Were you aware of those conclusions

at the time when you did your study, when you prepared your testimony?

A. (Witness Meyer) We have always been aware of the possibility. In NUREG 850 it is clearly indicated that if the determination of the specific Indian Point study was that there was no undue risk at Indian Point, that the mitigation study would not be then singled out for this particular facility, but would rather be folded into the more generic long term study of mitigation features for nuclear power plants.

Q. Were you aware specifically of what would be said in the Rowesome and Long testimony with regard to mitigation features at the time you prepared your testimony?

A. (Witness Meyer) I was not.

Q. In your work did you find the mitigation effects -- I am sorry, the risk reduction effects of the staff's mitigation package to be greater or less than you had previously expected prior to doing the computer runs to prepare the testimony?

A. (Witness Meyer) The risk reduction values were lower, that is the safety benefit was less significant, after performing the specific analyses, lower than we had first anticipated based on more general studies that you are familiar with.

Q. What had you anticipated based on those studies?

A. (Witness Meyer) Well, the studies, in particular studies performed at Sandia National Laboratory, indicated that the risk reduction potential from, say, the filtered vent system would be of the order of ten and higher.

We had no reason to believe otherwise, although we did know that the initial studies were less detailed than the ones that we ended up doing for the Indian Point site.

It was only until we had factored in the external event fixes and completed our analysis that we arrived at the values that we have presented here in this testimony.

Q. Now, by external event fixes you are referring to those that are covered in IPPSS amendment one, are you not?

A. (Witness Meyer) The fixes I am referring to are indicated on page 8 of our testimony. You will notice that the damage states are indicated there, the probabilities of the various damage states are indicated there, both before fix and after fix.

How well those fixes conform to the amendment one description from the utility, I don't know.

Q. Well, you are aware that in general amendment one does deal with reduced seismic fragility, reduced fire vulnerability, and anticipated shutdown for hurricanes, are you not?

A. (Witness Meyer) Yes, I am aware of that.

My point is that the determination of the change in the frequencies of damage states were provided to me by the staff, and I was not involved in whether or not that staff estimation of reduction was consistent with the utility amendment one submittal.

Q. Right. But to the best of your knowledge the data on which the fix was based was reviewed neither by yourself nor by any other member of the staff. Is that correct?

A. (Witness Meyer) All I can say is it was not reviewed by me.

Q. Thank you.

MR. BLUM: We have no further

questions.

JUDGE GLEASON: Do you have any redirect, Ms. Moore?

MS. MOORE: Could I withhold my redirect until everyone is finished, Your Honor?

JUDGE GLEASON: If you prefer.

MS. MOORE: I prefer.

JUDGE GLEASON: Mr. Brandenburg, or Mr. Colarulli?

CROSS EXAMINATION BY MR. BRANDENBURG:

Q. Gentlemen, I would first like to start with page 3 of your testimony, and you indicate there that you analyze plant damage states using the MARCH computer code.

Could you tell us which version of the MARCH code you employed?

A. (Witness Pratt) MARCH 1, part 1.

Q. Now, on the next page of your testimony, page 3 B 4, you state that you assumed that containment isolation would not occur, one in one thousand, that is ten to the minus three, times due to the fact that building penetrations were left open.

What was the basis for that assumption?

MS. MOORE: Your Honor, that's asked and answered.

JUDGE GLEASON: Let him answer.

A. (Witness Meyer) As I mentioned earlier, that number was provided to me by another group at NRC.

It was determined based on generic as well as specific characteristics of the Indian Point containment isolation capability, as I understand it.

Q. Are you aware, or were you provided information by those persons on the staff with whom you discussed this subject, the instrumentation available to the operators in the control room indicating the isolation status of the containment at the Indian Point plant?

A. (Witness Meyer) No, I was not.

Q. I would like to just ask you a few questions on the containment failure point assumption which you were asked about by Mr. Blum. I think it would be useful if you would turn to page 3 B 20 of your testimony.

Now, referring to the array of probability of failures set forth on that page, what would the effect on the damage states which you develop in your testimony be if 126 pounds p.s.i.g. was assumed to be the mean value of failure, rather than the 116 indicated?

A. (Witness Meyer) It would have two effects. The over pressurization and failure would, in terms of net effect, occur later in the accident, and several of the hydrogen burn failure modes assumed would no longer be present because the failure pressures would not exceed the higher containment failure pressure.

Q. And as a result of such an assumption the frequency of containment failure would be reduced. Is that correct?

A. (Witness Meyer) If the same 5 p.s.i. standard deviation value was maintained, yes, that's correct.

Q. Now I am going to ask you a slightly different question about the same approach, and that is what would be the effect on damage statements if 126 p.s.i.g. was assumed to be the lower bound, that is the minus 2 sigma, which in your testimony here you assume to be one of 6 p.s.i. I ask you to assume it's 126 p.s.i.g.?

A. (Witness Meyer) Well, it would be similar to the effect I just spoke to. The overpressurization failures would occur later, and we would have fewer hydrogen burn failures, the net result being a lower probability of containment failure.

Q. Now, the staff's analysis of the containment failure modes at Indian Point containment determined, did they not, that 126 p.s.i.g. is the onset of yielding, rather than the actual point of gross failure of containment. Is that correct?

A. (Witness Meyer) It's the point at which there is extensive yielding in the rebar. Yes.

Q. Now just, Dr. Meyer, to tie this up with some earlier testimony on this subject, I am correct, am I not, that 126 p.s.i.g. is the equivalent of 141 p.s.i.a.?

A. (Witness Meyer) That is correct.

Q. My next question relates to page 21 of the testimony, the parenthetical sentence starting on line 4 of that page, in particular.

Now, as I understand it, your evaluation assumes the presence of water in the cavity at the time of vessel failure for the L F damage state, and as I look at table 3 B 1, on page 3 B 9, I am informed that that damage state is one in which the fan coolers are assumed to be operational. Is that correct?

A. (Witness Meyer) That's correct.

Q. Now, returning to page 3 B 21, you go on to state that water might not be present for the EFC damage state.

And returning again to the table 3 B 1, I see that that damage state is one in which both the sprays and the fan coolers are operational.

My question is if water would be present under the LF damage state when the fan coolers are operational, why would similarly the water not be present when both the fan coolers and the sprays were operational?

A. (Witness Meyer) The L signifies a late core-melt. In the recirculation mode you have had the opportunity during the invection mode of injecting the total inventory of the refueling water storage tank into the containment system. This virtually guaranties a flooding of the reactor cast prior to vessel failure.

For the EFC, where we have an early melt, and failure of invection, the fans can keep the pressures low for a considerable period of time, thus not permitting the sprays to come on until later, so we are not guaranteed that substantial amounts of refueling water, storage tank water, has entered the containment. Therefore we didn't think that we would have the same guaranty of extensive cavity flooding at that particular damage state.

Perhaps Dr. Pratt would like to comment further on that.

A. (Witness Pratt) The only additional comment I would make is that at the point of the vessel failure, when we do get a large pressure rise, at that point the containment sprays are actuated, and they would be actuated as in Three Mile Island when there was a hydrogen burn.

But here we are talking about water being. It's a kind of an intermediate stage between an E damage state where a good deal of the water that is available is in the atmosphere in the form of steam. We need to take most of the available water inventory and put it into the containment building to build up the pressures.

For this case you would have a condensation of some of that water, so rather more water available than the E damage state, but, again, not the vast amount of water that is associated with the refueling and water storage tank.

Q. Now, at the Indian Point plants are you aware that any electrical cables inside containment are not necessary to maintain spray operability due to the fact that certain pumps that would maintain spray operability are located outside the containment?

A. (Witness Meyer) There are recirculation spray pumps located outside the containment. And I believe I mentioned that it was my perception that for the sprays, there very well may not be cables that would see a hydrogen burn.

Q. Now, in particular with the RHR and the residual heat removal pumps, are you aware that at Indian Point those pumps are located outside the containment?

A. (Witness Meyer) I am aware that one of the two sets of pumps are located outside the containment. I don't specifically remember whether it's the RHR pumps.

Q. And these pumps would be available to maintain operability of the sprays?

A. (Witness Meyer) They, yes, that's correct. And it has been pointed out quite correctly that that is a feature of the Indian Point containment that is very positive in that respect.

Q. The operation of these pumps outside of containment, and their maintenance of the sprays, would therefore be unaffected by considerations of hydrogen and combustion occurring inside containment. Is that correct?

A. (Witness Meyer) That's why I said that statement early year, that I felt the containment sprays would not be affected by hydrogen burns.

Q. With respect to the cables, themselves, turning to the equipment that is, indeed, inside the containment, are either of you aware of any tests that have been performed which demonstrate that the electrical cables ignite in the presence of hydrogen combustion when the concentration of hydrogen is in the 4 to 14 percent range that you mentioned earlier?

A. (Witness Pratt) I can't give you a reference on that.

The consultant that we had working in

this area did give me a report in which he noted some of these experiments. They were done, I believe with fracturing.

Q. And do you recall whether there was any demonstration of ignition of cables in the presence of hydrogen combustion in that concentrate range?

A. (Witness Pratt) Well, I don't believe they presented hydrogen combustion calculations. They subjected the cables to a thermal radiation field.

I might add that this was not over the -- I would have to get you the report. It was for a range of cable materials that he did his studies. Some materials were better than others at resisting the effect of ignition under these circumstances. I don't know what particular composite of materials there are at Indian Point containment facilities.

Q. Now I would like to turn to the subject of the amount of hydrogen that would be generated in the event of various assumed core-melt sequences.

As I understand your testimony, you have assumed that one hundred percent of the zirconium in the reactor vessel would be oxidized due interaction with the draining of the water and so forth. Is that correct?

A. (Witness Pratt) That's correct.

Q. And you made this assumption for all scenarios in which you were modeling core-melt sequences. Is that correct?

A. (Witness Pratt) Yes.

Q. Now, did you evaluate the different scenarios to ascertain whether some might be subject to greater oxidation than others?

A. (Witness Pratt) Yes, we did.

Q. And what was the basis for your decision then to assume complete zircloid oxidation in all scenarios?

A. (Witness Pratt) The assumption we talk about in the testimony of one hundred percent zircloid reaction really is a mix in terms of the amount of hydrogen we could imagine produced during the core-melt down progression within the reactor vessel.

We would include in that estimate of around two thousand pounds of hydrogen a certain factor coming from the external, a relatively small amount, around 10 percent. It could be rather loosely described in the testimony as a hundred percent zircloid reaction.

What we are really talking about is during a core-melt down event within the reactor vessel we see the potential being produced from both steel and oxidation reaction on the order of two thousand pounds of oxygen. We could then ---a thousand pounds of pressure being produced ex vessel by oxidation of the steel.

Q. But these were assumptions, is that not a fact? In other words, they were not your best estimates?

A. (Witness Pratt) No, they would be, in my opinion, the best estimates that we could do at this stage. There are estimates that are out in the literature that oxidize considerably more hydrogen, for example. We felt they were inappropriate. That is why we chose three thousand pounds.

Q. You used the MARCH 1.1 1 code to do this. Is that correct?

A. (Witness Pratt) Yes.

Q. Mr. Pratt, are you aware that the MARCH 1.1 code has been criticized for a number of assumptions which it makes, and superseded in many applications by the MARCH 2 code?

A. (Witness Pratt) Very much so. In fact, we were highly instrumental in criticizing March 1.1. I gave an extensive presentation in front of the ACRS on that. We participated in the Sandia review, and in addition to that we also provided a very large number of the modifications to the MARCH 1.1 that were provided to Patel Columbus.

At Brookhaven, and we are involved very heavily at present in a very heavy peer review of March 2. So I am very fully aware of the developments.

Q. Are you aware that the authors of the IPPSS study performed a physical assessment of the amount of hydrogen that would be generated according to the MARCH code, and reached the conclusion that with respect to the quantity of hydrogen that would be generated, that the March 1.1 code yielded erroneous results?

A. (Witness Pratt) I am, and in the testimony I believe we discuss in detail why I believe that that particular conclusion was, if you like, a limiting calculation.

This was brought up by Mr. Blum in his cross examination.

Q. Now, in connection with your testimony did you perform any independent analysis of the extent of oxidation that would occur? Did you do any modeling personally, or --

A. (Witness Pratt) Personally, no. People in my group at Brookhaven have done, and, indeed, some of the modifications that we did provide to Patel Columbus allowed for more gradual interactions between the core degree of water which resulted in rather more benign pressure rises, how it maintains to correspond materiality at a relatively hot temperature which allows oxidation to occur on a longer time frame.

The assumption that was made in IPPSS was that the convection results would be rather rapid, and as the temperature comes down rapidly, there would not be time -- so in some ways the models that we incorporated into the MARCH code did allow for longer times to oxidize the metal. They also resulted in longer periods before potential failure of the containment building from this model.

Q. Now, let's move on from the zircloid to the lower internals. On the top of page 3 B 13 you talk about the one hour in which the core materials would fall the bottom of the pit and cause a vessel failure.

Can you describe for us, and I don't mean to limit this to either of you, can you tell us how, under your modeling assumptions, the vessel bottom would fail?

A. (Witness Pratt) Yes. The calculations that is done, that is internal to the MARCH code, it looks at the loadings on the bottom of the reactor vessel in terms of the dead weight of the core materials, the internal pressure of the pressure vessel relative to the outside containment atmosphere. So there is a delta P effect.

And also the thermal degradation of the head, and we are talking about a gross degradation of the bottom of the vessel.

Q. You are aware that the authors of IPPSS assume that the initial failure of the vessel would occur at certain weld points and at certain instrument thimble entries, and so forth, in the lower vessel. Is that correct?

A. (Witness Pratt) Yes.

Q. In connection with your testimony did you evaluate the liklihood that such localized failures would initially occur?

A. (Witness Pratt) Only in a parametric fashion. In other words, we assumed one can tell the MARCH code to fail and the vessel had immediately, and do the calculation that way.

But, indeed, for certain high pressure cases, for the E damage state, where we don't induce the failure of the coolant pumps, for instance, where we maintain the primary pressure high, then even the MARCH code predicts a very rapid failure of the vessel under those circumstances.

Q. If the localized failure of the vessel as modeled in IPPSS were to occur, what effect would that have on the amount of hydrogen contribution from the steel, itself?

A. (Witness Pratt) It would reduce the amount of steel that was available in the reactor cavity to oxidize and produce hydrogen, simply because you are not melting the entire bottom of the reactor vessel. That mass is about fifty thousand pounds of steel.

Q. Mr. Pratt, would the rapid failure that you refer to in your most recent answer affect the amount of hydrogen generated in the vessel, do you recall?

A. (Witness Pratt) In the vessel?

Q. Yes. In --

A. (Witness Pratt) Yes, it would reduce the amount of steel available that the core material would be in touch with the water in the bottom of the vessel.

Q. With the result of lesser generation of hydrogen, is that correct, in the vessel?

A. (Witness Pratt) Yes.

Q. Gentlemen, let's turn to the subject of water, if we may, and while we are on page 3 B 13 I would like to point you specifically to the last sentence in the first full paragraph where you state, "However, there is not sufficient water available in the cavity to bring the core debris into thermal equilibrium."

And this is a subject that you revisit in much more detail, I guess starting on page 3 B 52 and thereafter.

My first question is, assuming the full quantity of core materials in the cavity -- well, let's get some numbers out here.

How many pounds of uranium materials would you expect to find in the cavity assuming a complete melt, approximately? And my next question will be how much water?

A. (Witness Pratt) I know exactly the amount of water. It's 200 thousand pounds of water.

Q. So your analysis, let me include, then, that 200 thousand pounds of water would be needed to quench the core. Is that essentially correct?

A. (Witness Pratt) That's correct. And that corresponds to the total amount of water in the accumulator tanks.

Q. Now, prior to the addition of the accumulator water, how much water did you calculate would be available merely from the primary system, by itself?

A. (Witness Pratt) This we dealt with somewhat parametrically. We were not sure how much water would be held up in the sumps and the floor. There is a lip around the entrance into the reactor cavity and there is a potential seen for holding water in the sump and a considerable amount of water on the floor.

The way we looked at the calculation, and this is something one can do by hand, which makes it rather reassuring, is to take all of the water available in the primary system, about six hundred thousand pounds of water, without 200 thousand pounds of water, and calculate the amount of steam that is necessary in the containment building to reach 140 p.s.i., and, lo and behold the two just about balance out.

So we felt that it was something of a stretch of imagination to assume that all of that water would get down into the cavity to mix with the core debris to get back up into the building. That's why we came to the assumption that for these E damage states there would be limited water available.

Now, the IPPSS assumes that we run out of water about the time of containment failure.

My thought is it would most likely be held up in the sump so that's really why we end up with a different assessment of the amount of water that might be available to these damage states.

Q. Now, we can agree, can we not, Mr. Pratt, that if you have the total amount of water in the primary system, plus the temperature pressure of the environment, that it's a rather easy calculation to apportion that water between steam on the one hand and liquid that would be available to the cavity on the other hand. Is that generally correct?

A. (Witness Pratt) I think that's what I just said.

Q. All right.

Now, at the point in your analysis in which you were addressing whether or not there would be quenching of core materials at that point in the progression of the accident sequence, what assumptions did you make as to the temperature and pressure that would be present in the containment environment?

A. (Witness Pratt) At this time there is around about -- that's given on page B 12. We are talking around about 60 p.s.i.

Q. And what is the temperature in the containment environment?

A. (Witness Pratt) It would be at saturation corresponding to 60 p.s.i.

Q. Now, let's move on from the mere availability of the primary system water, and look at the accumulated water.

Under what circumstances did you assume that the accumulator water would be available?

A. (Witness Pratt) Under all circumstances.

Q. And when you added that water to the amount of water that was available from the primary system, did you exceed the 200 thousand pound threshold that you referred to at that point?

A. (Witness Pratt) I guess I am confused at your question. Could you repeat that?

Q. Well, when one assumes the presence of both the primary system water and the accumulator water do you exceed or not exceed the 200 thousand pounds of water which you indicated was required to quench the core?

A. (Witness Pratt) Well, it's sequence dependent. As I said before, you require about 200 thousand pounds of water to quench the core debris. That corresponds exactly -- not exactly, actually the quantities of about 170 thousand pounds in the accumulators, and you really need but 200 thousand pound to quench. You get 6 hundred thousand pounds from the primary system, so yes, there is more than enough water in the primary system, assuming it's available in the cavity, to completely quench the core.

Q. Let's turn to page 3 B 20 of your

testimony.

In light of what you just said, Mr. Pratt, could you explain to us why, under the E F version, which if we turn to table 3 B 1 we see is early core-melt fan coolers operational, that under that damage state a split fraction for the quenching was .012?

A. (Witness Pratt) Well, I think this is one of these split fractions that if we had it to do over again we would have changed it. It's not a very large impact. We did a very detailed sensitivity study looking at the impact of that, and it isn't large.

Going by the discussion that I have just given you, I think we would go by, and I would believe, the E volume, which is .01, and the EF would be somewhere in between, probably more appropriately somewhere around 0.5.

This was the first test we did, and in going through it in terms of the water available a better estimate would have been .5. We did that calculation with .5 and found there would be absolutely no impact at all.

Q. Now, given a more recent analysis if the water would be available, I would like to ask you the same question with respect to the EFCC sequence listed.

Only 3 B 1 is an early core-melt damage state in which both the fans and sprays are operational. And you use a frequency of.9.

Would your answer be the same generally as with respect to the EF sequence?

A. (Witness Pratt) Yes, that would be true. As I say, we have put these things in the containment of entries, and one of the nice things about it was there was not the sensitivity to risk that one would expect.

Q. And these split fractions referred to on page 3 B 20 of your testimony were developed using the parametric analysis. Is that right?

A. (Witness Pratt) No, they were based, I believe, upon the type of discussion we have been having now. From our knowledge of the amount of water that is available, from the types of sequences that are going into these damage states, we came up with these split fractions. We can vary these values around parametrically, and we do that, and we get very little impact on risk as a result of it.

Q. Now, in those damage states where the fan coolers are operational would you agree that given the configuration of the Indian Point containment that if there were debris in the cavity, and the fan cooler units were operational, that water would be continually supplied to the containment floor from condensation from the fan cooler units, and that that water would spill into the cavity?

A. (Witness Pratt) I think there is a reasonably good potential for that. Certainly not as good a potential as for the EFC damage state, where you would have refueling water storage tank water also into the cavity.

Q. Now, in those scenarios where the refueling water storage water contribution was not assumed to be present, and just looking at those scenarios for the moment, in your analysis did you perform any sort of mass balance analysis of the delivery and distribution of the water? I am talking now only about the primary system and the accumulator water.

A. (Witness Pratt) Yes. As part of MARCH analysis, the MARCH code calculates the amount of condensate that is available and then how much could be held up in sump, and the remaining water would be relocated into the reactor cavity.

Doing that type of analysis you can get an idea of how much water is available. But it's a matter of how much you assume will be held up in the sump.

Q. Now let's turn to the scenarios where the contribution from the refueling water storage tank would, in fact, be available.

Of the various scenarios which you evaluated in your testimony, when would the refueling water storage contribution not be available? That is in what situations will you have only the accumulator water on the primary system water?

A. (Witness Pratt) For the E damage states and for the damage states EF. All of the sequences would eventually have refueling water storage tank water.

Q. Well, in the EF state, where the fan coolers would be operational, I guess I was unclear as to your answer, Mr. Pratt, about the delivery of condensation from the containment atmosphere from the fan coolers to the cavity.

Could you try once more to explain to us what assumptions you are prepared to make about the delivery of water to the cavity from condensation from the fan coolers?

A. (Witness Pratt) I think that I said that there is a good potential for the recycling of the condensated water into the cavity under these circumstances.

Q. Now, let's move to those situations where we do have debris in the cavity, and we do have sufficient water from whatever source over the debris.

What assumptions did you then make about the coolability of the debris bed in your testimony?

A. (Witness Pratt) If it's flooded we would assume that there is about a 90 percent chance that it would be coolable, and that there would be a 10 percent chance that there would be a basemat penetration.

JUDGE PARIS: I didn't hear that.

A. If it's flooded we would assume that there is about a 90 percent chance that it would be coolable, and that there would be a 10 percent chance that there would be a basemat penetration.

Q. Now, what would be the phenomenology of the 10 percent, if you will? Under what circumstances did you presume that although water was present, that, nonetheless, the debris may be not coolable, and there would be a concrete attack?

A. (Witness Pratt) There is a very detailed discussion on this in NUREG 850. There is experimental evidence, albeit on a small scale, that indicates core material might pass into the cavity, not mix with the water, that there might be a crust formed between the core debris and the water, that these crusts are porous, and that gases released from the concrete could be released through the crusts into the atmosphere, and there would be essentially a separation of the molten core material.

Now, these were done in Germany by Pease, I believe. That is assuming the water is having no effect on the core material.

Consultants to the NRC felt that the stability of crust over a large area was in question, and certainly on the scale that we are talking now. So there was a feeling that this may relate to the transition phase, but that the crust would break up and water would eventually get in and cool it. And that's why during this transition phase we allowed about half an hour for the core material to cool than calculated, and that's where the additional thousand pounds of hydrogen came from.

If we assumed, as was done in the IPPSS, that the cooling occurred very rapidly, then we would get much higher pressurization, but very little hydrogen produced.

Q. Do you recall what assumptions the IPPSS study used?

A. (Witness Pratt) I think it was .9999. Four nines, I believe.

Q. Now, just to give us some frame of reference on this crust --

A. (Witness Pratt) That's my recollection. I could be wrong.

Q. Just to give us to frame of reference on this crust phenomenon that would lead to possible noncoolability, notwithstanding the presence of water, Mr. Pratt, what was the size of the crust that was modeled in the German studies that you referred to?

A. (Witness Pratt) You are getting into rather detailed assessment.

Q. Well, I am driving at the translatability of the small scale experimental data that you refer to in the 10 percent assumption that you made in your testimony, that notwithstanding the presence of water there might not be a cooling. Could we have some appreciation for how translatable these small scale tests would be for a full scale situation?

A. (Witness Pratt) I think I already testified that because of our concerns regarding crustability, that's why we gave a relatively high percent, 90 percent, they would be coolable.

Q. What would be the various reactions that would occur that would tend to break up any crust? Would the release of oxygen from the concrete reaction tend to --

A. Well, the gases are CO2, water, and they are reduced as they pass through the melt to hydrogen and CO.

We have at the Indian Point facility basalt concrete, so there is little CO produced, so we are talking about primarily water which is reduced to hydrogen.

Yes, the blowing rates certainly would assist the breaking up the crusts.

Also simply the spans over which the crust is existing would tend to break it up also, or would have to exist.

Q. Gentlemen, I would like to ask you a few questions about page 3 B 33 of your testimony

JUDGE GLEASON: Mr. Brandenburg, how much more cross examining do you have?

MR. BRANDENBURG: There is considerably more, Mr. Chairman. We have to get on to the new medication features discussion.

JUDGE GLEASON: I suggest we take a ten minute break.

MR. BRANDENBURG: Thank you, Mr. Chairman.

(There was a short recess.)

JUDGE GLEASON: If we could proceed, please?

JUDGE GLEASON: Go ahead, Mr. Brandenburg.

MR. BRANDENBURG: Thank you, Mr. Chairman.

Q. Gentlemen, before our previous recess I was about to ask you a series of questions relating to pages 3 B 32 through 35 of your testimony in which you model the E, EFC, OF and EF damage states.

Let's start with page 3 B 32, which

is the E accident sequence or damage state, and that is one in which there is an assumed early core-melt with no containment cooling. And my question is a when question, not the whether question.

At what point in time in the progression of the accident sequence do you assume that there would be a breach of containment for this sequence?

A. (Witness Meyer) For the over pressure indication failure mode?

Q. Yes. Damage state E, which is early core-melt with no containment cooling.

A. (Witness Meyer) For the over pressure indication we have used the value of 13 hours time to containment failure.

Q. That's 13 hours counting from what?

A. From the scramble of the reactor starting the accident.

Q. Now, turning to page 3 B 33, this is the accident sequence EFC which you described as early core-melt with both sprays and coolers operational.

I understand from my review of page 3 B 33 that you have a no fail determination of approximately 80 percent. And my question relates to those 17 percent of the instances in which there is a failure by basemat penetration, and my question is when do you assume that would occur, how many hours after the start of the accident sequence?

A. (Witness Meyer) Three days.

Q. And how about the over pressure indication category F, on page 3 B 33, which, as I understand it is an over pressure indication failure to which you accord a 3 percent liklihood?

A. (Witness Meyer) This is release category F, and if you turn to page 25 the estimated release time for that, failure time for that, is three hours.

Q. All right. Maybe we can save some time.

These various release categories E and H are the release times all as shown on page 3 B 25 independent of whether the sequence you are concerned with is an EFC sequence or a F sequence or an EF sequence?

A. (Witness Meyer) Could you repeat that question again, please?

Q. Well, we finished with the E sequence, which was 3 B 32. We are now at the EF sequence, and I think we have established that the category F release, the time of release would be as shown on page 3 B 25. And I think you indicated that the basemat failure mode which is category H was three days, and I see that also on page 3 B 25.

My question is is the H category failures for the remaining sequences, sequences LF, as shown on page 3 B 34, and sequences EF as shown on page 3 B 35, similarly the times shown on page 3 B 25 for the category H release?

A. (Witness Meyer) That is correct.

Q. So, in effect, your analysis, the time of release, as long as you are concerned with the category H basemat type release, that the timing of that failure is independent of whether or not you are dealing with a LF sequence or a EF sequence, things of that sort. Is that correct?

A. (Witness Meyer) Yes. The period of time was sufficiently long that going into any details regarding variations on that three days was considered to have negligible importance.

Q. Now turning to the EF sequence as shown on page 3 B 33, and I suspect this is a question for you, Mr. Pratt, on the basemat failure category H in the sequence, if I understand page 3 B 33, your analysis accords that in the before fix mode as having a frequency of approximately 17 percent.

What is the phenomenology that would lead to the 17 percent assumed frequency of basemat failure, given the assumption that both the sprays and coolers were operational? Is this this encrustation phenomenon that we were discussing before the break?

A. (Witness Pratt) You really have to go back to mode 4 on that graph, and look down and see that for 10 percent of the time you assume the cavity could be dry. And we assume there would be a 90 percent chance of basemat failure for that sequence. You have to follow through the lower branch of that tree.

If you assume the cavity is flooded, going in the upper direction. Then you have a 90 percent of no fail and a 10 percent chance of a absemat failure.

The net result is when you add the two together you come up with about 17 percent. Half of that is coming from the assumption that there would be some emptied water in the cavity for that case

Is that clear?

Q. I think so.

Now, turning to page 3 B 34, which is sequence LF, which, as I view table 3 B 1 is late core-melt with fan coolers operational, you have a basemat failure occurring approximately 9 or 10 percent of the time.

Can you tell us what the considerations were that lead to basemat failure under those circumstances?

A. (Witness Pratt) Again I think I am repeating myself. If you go to note 4, which assumes the cavity is flooded, we went through that.

For that case we then assume that 10 percent of the time there would be a potential for basemat failure, even though the cavity was flooded and the core debris was flooded, and that's where the 10 percent comes from.

Q. Just to hit the last table, which is 3 B 35, where, in the EF sequence, which is the early melt with only the fan coolers operational, your analysis indicates a basemat penetration occurring approximately 79 percent of the time, what are the considerations that contribute to that?

A. (Witness Pratt) Again, this is the sequence that we discussed earlier, where perhaps we would have changed. We have a 99 percent chance at note 4 of a dry reactor cavity and a one percent chance only of it being flooded.

We would have to give that a 50-50 percent chance now, of the best estimates of flooding of the cavity, so that would bring down the basemat failure mode by that ratio, by about a factor of 2.

As I said, we have made that calculation and made those changes and have found that it does not influence risk.

One further point. I may have said it correctly, I may not in the testimony, but I would like to clarify it for the record.

You specifically asked me for which damage states do we assume that the refueling tank storage water never goes into the containment cavity. It is for damage states E and EF. Somebody thought they heard me say LF. It is E and EF. Those are the only two damage states.

Q. Now, would it be fair to say for

those sequences in which you model basemat penetration, the frequency of the basemat penetration is to a large extent due to this lack of debris coolability which we discussed earlier?

A. (Witness Pratt) Yes.

Q. And that, in turn, the assumptions that one makes about debris bed coolability are dependent on how one models this encrustation phenomenon that you mentioned earlier. Is that correct?

A. (Witness Pratt) Yes.

Q. Gentlemen, one question before we move onto the mitigation strategy, and this relates to the assumption about the containment leak rate, and this appears on page 3 B 5 of your testimony.

Looking at the last paragraph on that page, I understand that you assume a heat rate of one percent per day. My question is are you aware of what the leak rate is that is set forth in the FSAR of the Indian Point units?

A. (Witness Meyer) The leak rate, I believe, is 0.1 percent per day.

Q. And why did you, in connection with your analysis, elect to increase that leak rate by a factor of ten?

A. (Witness Meyer) Because we were, on virtually all of these accident sequences, well beyond the design basis in many respects, and we thought it prudent to increase the leakage rate by an order of magnitude. We checked out the impact of that on risk, and as I think has been made clear in the testimony, even at that one percent per day leak rate, the impact on risk is negligible.

Q. Now, moving on to the subject of the mitigation features, one initial question to put this in context. Do either of you gentlemen know a Mr. Robert Benaro?

A. (Witness Meyer) Yes, I do.

Q. Mr. Pratt?

A. (Witness Pratt) Yes.

Q. Are you aware that Mr. Bernaro will be testifying next in this proceeding?

A. (Witness Meyer) Yes, I am aware that he will testify after us.

Q. You are aware, are you not, that the subject of the appropriate source curves to use in evaluating the risk at the Indian Point units has been an issue in this proceeding?

A. (Witness Meyer) Yes, I am aware of that.

Q. Now, the modeling of mitigation features which you make in your testimony makes what assumptions about source terms? I am not seeking detail precision here, but just some general description of the source terms that are presumed in your analyses?

A. (Witness Meyer) We used the same basic approach as WASH 1400.

Q. Now, would you agree that to the extent one were to use lower source terms, that is to say assume that the amount, frequency, and mix of radio neucleides that would be released to the environment in the event of a severe accident, would be less, that such assumptions would similarly reduce the potential benefit from any mitigative features?

A. (Witness Meyer) I could not draw that general conclusion at this time.

Q. In what situations would that not be an appropriate conclusion?

A. In what situations?

Q. Under what circumstances?

JUDGE PARIS: Excuse me. I am not

sure the witness understands the question. You said would not be, you said would be.

Q. You used WASH 1400 source terms, you assume a given quantity of radioneucleide release, and you then go on to evaluate several devices that in case of a filtered vent would diminish the release, in the case of certain other devices would either preclude a failure and release or substantially delay a release in terms of the point in time it occurs.

My question is to the extent that one would reduce the source term, that is the amount of radioneucleides that one assumes would be released, would that not similarly reduce the potential for any of the mitigative features?

A. (Witness Meyer) Each individual accident sequence would have to be considered separately.

I would not state that it would diminish the effective mitigation features if one considers mitigation of early containment failure.

On the other hand, the later the containment failure, the less attractive these mitigation features will become in terms of risk reduction.

Q. If one lowers the source term?
A. (Witness Meyer) That's correct.
Q. Now, a reduced source term, then, would diminish the potential benefits flowing from the heat pipes and the flooded cavity features, those both being ones that address the latent overpressurization failure primarily; is that correct, Dr. Meyer?
A. (Witness Meyer) That's basically a correct statement, yes.
Q. Now, when you were, Dr. Meyer, appearing before the Board some weeks ago, we discussed the so-called UCLA study of the mitigative features which you address in your testimony. I believe that's NUREG/CR-1666.
A. (Witness Meyer) That's correct.
Q. Since that time, a copy of that document has been supplied to the parties. You are familiar with that document, aren't you?
A. (Witness Meyer) Yes, I am.
Q. Now, if I understood your answers to some of Mr. Blum's questions earlier, subsequent to the time that this NUREG was prepared and subsequent to the time that you prepared your testimony that's before the Board at this time,

there was an evaluation by the NRC Staff of whether or not such mitigative features should be recommended for installation in the Indian Point proceeding; is that correct?

A. (Witness Meyer) Yes, that's correct..

Q. Now, were you a party to that process?

A. (Witness Meyer) Yes, I was.

Q. Now, turning to the UCLA study and, in particular, it's discussion of the heat pipes, perhaps, just one brief introductory question: Could you describe for us the basic purpose and construction and function of these pipes?

A. (Witness Meyer) Yes. The heat pipe is a closed-pipe system that contains a working fluid. It can be thought of as being in three parts. The part inside the containment is the evaporator portion, evaporating the working fluid and thus removing heat from the containment by condensation of water vapors on the outside surfaces of the heat pipe.

This vapors is then driven by a pressure differentiation through the transition portion which would go through the containment wall and out into a condenser portion where the vapor is condensed to water, dumping the heat out into the atmosphere.

The water then, either through a gravity feed or through a wick arrangement flows back down into the evaporator portion inside the containment.

By this process, heat is removed from the containment by a system that's completely sealed and is completely passive.

Q. Now, just so we are clear, Dr. Meyer, this particular system is among those mitigative systems which the Staff included not to recommend for installation to the Indian Point Units; is that correct?

A. (Witness Meyer) That is correct, yes.

Q. Now, have such heat pipe devices ever been tested in any prototype testing or design demonstration program?

A. (Witness Meyer) In connection with what application?

Q. In connection with their use in a nuclear power plant.

A. (Witness Meyer) Not to my knowledge.

Q. Would you think it would be fair to characterize the discussion of heat pipes contained in the UCLA study as a preliminary investigation?

A. (Witness Meyer) I think that would be a fair, general comment, yes.

Q. Now, given the requirements of heat transference, in order for such devices to be effective in a presumed postcoremelt environment, what would be the -- well, let me ask it more directly.

What was the wall thickness of the heat pipes that was evaluated in the UCLA study?

A. The UCLA study, I believe, had a heat pipe thickness of three millimeters.

Q. Now, based upon your knowledge of such devices and nuclear technology, would you think it fair to say that there would be major design problems that would be encountered in trying to seismically qualify pipes with three millimeters' thickness?

A. (Witness Meyer) I don't --

Q. And extending some hundred feet out into the outside of the containment and so forth?

A. (Witness Meyer) I don't think there's any question but that there would be some engineering challenges to produce such a system for practical application to this particular containment design.

Q. Would there similarly be a major engineering challenge, to use your phraseology, to qualify such devices for both internal and external missiles?

A. (Witness Meyer) They would, of course, have to meet the requirements of any device associated with the containment, and it is because of considerations like this that we considered such a device to be a very expensive one, and that was part of the reason for the decision as related in the Question 5 testimony for eliminating heat pipe as a possible option at this time.

Q. Now, the UCLA study assumed, did it not, that it might be possible, and I emphasize the word "might," might be possible to install such devices at Indian Point without interfering with the rebar in the containment structure; is that correct?

In other words, they held out some possibility, after review of the drawings of the rebar in the containment structure, that it might be possible to avoid piercing the rebar in installing such pipes. Do you recall that?

A. The UCLA study came to such a conclusion, yes.

Q. Now, would you consider it a further substantial disadvantage of attempting to retrofit such devices at the Indian Point Plants if it became apparent that it was necessary to interfere with the existing rebar in the containment structure in order to install such devices?

A. (Witness Meyer) Yes. This would be a major consideration.

It is my personal view that if you would take the approach of actually drilling new holes in the containment structure, that you couldn't avoid cutting through some of the rebar, and any analysis of its effect would have to take that into consideration.

Another consideration or another option that the UCLA study considered was making use of existing penetrations. Actually cutting holes in the containment wall was one of a number of options.

Q. One last question there and I think I'm through, Mr. Chairman: Do you recall how many layers or separate planes of rebar there are in the Indian Point containment structures?

A. (Witness Meyer) I don't have a number for you. There are extensive planes of layer upon layer of rebar that would have to be cut through in order to put holes in the containment.

Q. And another of the devices you considered was a system to insure greater frequency of cavity flooding; am I correct?

A. (Witness Meyer) Yes.

Q. That was the third --

A. (Witness Meyer) That's correct.

Q. Now, you concluded, did you not, that installation of that system, that is the cavity flooding system, if I can call it that, would be counterproductive if it were not installed in conjunction with the heat pipe system; is that correct?

A. Yes.

Q. You couldn't install one without the other. You had to install them both to get any kind of benefit; is that right?

A. (Witness Meyer) The requirement would have to be accompanied with a requirement for containment heat removal. In that sense, yes, you are correct.

MR. BRANDENBURG: Mr. Chairman, I

have no further questions of these witnesses.

JUDGE GLEASON: Mr. Colarulli, cross-examination?

CROSS-EXAMINATION

BY MR. COLARULLI:.

Q. Dr. Meyer, I'd like to follow up on just a couple of points that were referred to in your exchange with Mr. Brandenburg.

Would you turn to page III-B.50 of your testimony.

Now, the table at the bottom of page III-B.50, as I understand it, is a table that you called a "Realistic Assessment Of The Reduction That Would Be Provided By Heat Pipes"; is that correct?

A. (Witness Meyer) Yes. This is our estimate.

Q. Now, do these reduction factors reflect the several seismic considerations that you just made reference to, specifically the impact if one had to cut through the rebar containment and the problems associated with seismically qualifying the heat pipes?

A. (Witness Meyer) Those matters were taken into account only in the way that we did through the unreliability number. The unreliability number of five percent is considered large by most standards. We did not consider it explicitly, but we would anticipate it could be included in that five percent unreliability number.

Q. But isn't it true that your position is that further analysis is needed to actually realistically model in and calculate the impact that would occur given the addition of seismic qualification?

A. (Witness Meyer) That's correct. The analysis in front of you gives an indication of the impact of unreliability and attendant risks on the reduction in the risk reduction afforded by these mitigation features.

Further analysis may reduce that unreliability number or it may increase that unreliability number.

Q. One question concerning the glow plug igniters that you referred to on page III.B-29 of your testimony. Your testimony does not reflect the specific risk reduction opted by glow plugs alone, doesn't it?

A. (Witness Meyer) No, we did not include the separate risk reduction that would result by having glow plugs alone installed in the containments.

Q. Have you made such a calculation?

A. (Witness Meyer) Yes, we have.

Q. And what results did that calculation yield?

A. (Witness Meyer) Let me turn it over to Dr. Pratt who has the exact numbers.

A. (Witness Pratt) It would have negligible effect in terms of acute fatalities for Units 2 and 3. It would have about 10 percent impact on latent effects for Unit 2, and for Unit 3 for latent effects, it would have negligible impact.

Q. Is it true that two possible attendant risks of glow plugs include an increase in containment pressure and some potential adverse equipment burns?

A. (Witness Meyer) The whole purpose of glow plugs is to burn hydrogen, and one would expect that you would burn lean mixtures of hydrogen with the glow plugs installed.

If they were not installed, you would expect more vigorous burns that would have greater effects on the devices you just mentioned.

So although you could say that it is an attendant risk that is considered, you are avoiding a much more damaging problem by burning early and burning your hydrogen lean.

JUDGE SHON: Dr. Meyer, your answer there and particularly the last part of your answer in which you mentioned "burning early" makes, it seems, some sort of an assumption to the effect that hydrogen mixtures per force must grow from lean mixtures to rich mixtures and that there must be a stage in which the glow plug would be ignited lean.

It would seem there would be some scenarios, at least in some places, in which a rich hydrogen mixture could develop and then go wandering to find a dull plug.

Wouldn't that put things in a bad state?

MR. MEYER: There are situations. For example, certain vessel failure situations where large amounts of hydrogen would be released at one time, and you would not have this lean mix that is more associated with core degradation than coremelt; and vessel failure and under those circumstances, a glow plug could have an attendant risk of igniting a relatively rich mixture of hydrogen.

JUDGE SHON: It would seem as if it could, perhaps, even ignite a rich mixture under circumstances where, left to itself, the mixture would lean out by mixing with the rest of the atmosphere and become less of a hazard, could it not?

MR. MEYER: Yes. This would have to be carefully considered in the placement of glow plugs in the containment.

JUDGE SHON: Thank you. I'm sorry for interrupting.

MR. COLARULLI: That's fine. I have no further questions.

MS. MOORE: Might I have a moment?

JUDGE GLEASON: Yes.

MS. MOORE: The Staff has no redirect.

JUDGE GLEASON: All right. Gentlemen, you are excused. Thank you very much.

Call your next witness, Miss Moore.

MS. MOORE: Your Honor, the Staff calls to the stand Mr. Robert M. Bernero.

JUDGE GLEASON: Miss Moore --

MS. MOORE: Your Honor, we are

trying to locate a copy of Mr. Bernero's testimony for the court reporter.

JUDGE GLEASON: Has he been sworn in previously?

MS. MOORE: No, he hasn't.

JUDGE GLEASON: Would you please stand and raise your right hand.

Whereupon,

ROBERT M. BERNERO

was sworn in by the Administrative Law Judge and testified as follows:

DIRECT EXAMINATION

BY MS. MOORE:

Q. Mr. Bernero, would you please state your name and business address for the record.

A. My name is Robert M. Bernero. My business address is US Nuclear Regulatory Commission, Washington, D.C.

Q. Would you please state your position with the Nuclear Regulatory Commission?

A. I am the director of the Accident Source Term Program Office in the Office of Nuclear Regulatory Research.

Q. Do you have before you a copy of the document entitled "Testimony of Robert M. Bernero on Severe Accident Source Terms"?

A. I do.

Q. Was this testimony prepared by you or did you participate in its preparation?

A. Yes, it was prepared by me and with my participation.

Q. Do you have any additions or correction to this testimony?

A. Yes, I do. I have one change to read into the record.

On page five, just below the middle of the page is a line which reads: "Failure of containment within 30 minutes of the initiation of the accident," and the words "30 minutes" should be replaced by the words "a few hours".

Q. With this change to your testimony, is it true and correct to the best of your knowledge, information and belief?

A. Yes, it is.

Q. Do you adopt this as your testimony in this proceeding?

A. Yes, I do.

MS. MOORE: Copies of this testimony have been delivered to the Board, the parties and the court reporter. I ask that the testimony and the attached professional qualification of Mr. Bernero be received into evidence and bound into the record as though read.

JUDGE GLEASON: Objection?

Hearing none, the testimony of Mr. Bernero will be received into evidence and bound into the record as if read.

(The bound testimony is as follows:)

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

TESTIMONY OF ROBERT M. BERNERO
ON SEVERE ACCIDENT SOURCE TERMS

Q.1 Please state your name and position with the NRC.

A.1 My name is Robert M. Bernero. I am the Director of the Accident Source Term Program Office in the Office of Nuclear Regulatory Research at the U.S. Nuclear Regulatory Commission.

Q.2 What are your responsibilities in that position?

A.2 My responsibilities in this position are to assure that Source Term related research results are implemented in policy and regulatory practice in a timely manner.

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

A.3 Yes, the statement of my professional qualifications is attached to this testimony.

Q.4 What is the purpose of your testimony?

A.4 This testimony provides the Staff's assessment of the status of current research in the subject matter raised in the testimony of Drs. Stratton and Rodger, and Mr. Potter, bound into the record following T.R. 8169, i.e., the quantity and characteristics of the radionuclides released following an accident, often referred to as the radiological "source term." (hereinafter referred to as the Stratton testimony)

Q.5 How is an assessment of the Source Term conducted?

A.5 The words "source term" are often used as a simple term suggesting that severe (severe core damage or core melt) accidents in nuclear reactors can be characterized by a single source term; and one might speak of reducing that source term by some factor, say of 10 or 5 or 100. In fact, each severe accident in a nuclear reactor of a specific design has a characteristic scenario and, therefore, has a characteristic pathway by which the radioactive material is moved from the core through the reactor coolant system out into the containment or other buildings and ultimately out into the biosphere. Each severe accident scenario then has a characteristic source term describing the amount and form of the radioactive materials which are released to the environment. In order to assess radiological source terms comprehensively, one must examine each of these accident scenarios and study the behavior of each of the principal species of radionuclides in each phase of these scenarios. Thus, a rather complex assessment is made, there are many source terms, and many complicated potential changes in source terms.

Q.6 Please provide the Staff's assessment of the Source Term questions and research activities discussed in the Stratton Testimony.

A.6 The Stratton testimony draws attention to the intense research program currently underway which is intended to resolve some of the questions concerning accident source terms identified in part by Dr. Stratton. We agree with the Stratton testimony's assessment that the accident source term methodology employed in the Reactor Safety Study (WASH-1400) results in conservative predictions, i.e., in overestimates of the quantity of fission products released to the environment. To the extent that the RSS methodology was employed, therefore, such conservatisms are incorporated in the Indian Point Probabilistic Safety Study (IPPSS) and in the staff's consequence estimates as embodied in the testimony of staff witnesses Meyer and Acharya. This agreement with Stratton et al. is based on the existing knowledge that several physical processes which would affect some degree of retention of radionuclides in various parts of the damaged facility were neglected in the RSS methodology for assessing radionuclide release and transport. The Stratton testimony draws heavily on quotations from NRC-sponsored research reports to establish this fact, so that repetition herein is not necessary. However, at the present time, we cannot agree with the next step in the Stratton testimony, i.e., the subjective estimates of quantitative "reduction factors" asserted on the basis of a qualitative description of a partial list of the physical processes affecting radionuclide release and transport. These estimates of reduction factors to be applied to the IPPSS are based solely on the subjective judgment of the authors of the Stratton testimony, and do not necessarily follow from the common ground of the general technical consensus described above. Although the current research effort may provide substantiation for some of the assertions of the Stratton testimony, the conclusions in the Stratton testimony regarding revised source terms are premature at this time. The staff does believe that the technical data available today strongly suggest that the WASH-1400 models for radionuclide behavior in severe accidents overestimate the amount of such material that would be released to the environment in such an accident. As I will explain later, the staff and its contractors are engaged in a substantial research program to obtain better data and models for estimating these source terms. It is our involvement in this work and the complexity of the analyses at this time that prompt us to say that selecting and using a new source term model at this time is premature.

The following examples demonstrate the reasons for the staff's inability to endorse the conclusions of the Stratton testimony.

1. The TMI Experience

A major factor influencing the conclusions in the Stratton testimony is the interpretation given to the TMI accident. The Stratton testimony (p. 15ff) sets up a comparison of the TMI event with release categories PWR-1 and -2, and in Table 4 (p. 53) with a WASH-1400 "average" and "lowest" release. On p. 17 the Stratton testimony notes agreement between the releases from the core for PWR-1 and -2 and the TMI observation, and then notes the lack of agreement of the releases from the containment. However, it should be noted that the RSS calculations are for accidents involving containment failure, while no failure of containment occurred at TMI. The PWR-1 and -2 release categories of the RSS are dominated by accident sequences involving catastrophic failure of containment within ~~30 minutes~~ a few hours of the initiation of the accident. In the TMI-2 accident, the principal releases were through the letdown and makeup system and the auxiliary building vent header many hours, even days after the accident. A direct comparison of the atmospheric releases from postulated events such as PWR-1 and PWR-2 release accidents with the TMI-2 accident where the atmospheric leakage from the containment was negligible cannot provide much insight concerning the effects of fission product transport behavior. This is not to say that valuable insight cannot be obtained from the TMI experience. Careful examination of the TMI data is expected to provide a useful reference for some mechanistic fission product release and transport models. However, some basic work, i.e., the examination of the TMI core, and determination of its fission product inventory and distribution is still to be completed. Completion of the TMI core examination is perhaps 2 years away.

2. Cesium Iodide

While the Stratton testimony almost quotes (Stratton, p. 22) the staff's conclusion that cesium iodide is the most likely predominant form of iodine released from the reactor core (NUREG-0772), Stratton concludes, without further quantification of the physical/chemical processes involved, that this change from the RSS assumption permits reduction of the quantity of iodine released to the environs. A factor apparently weighing heavily in this assessment is the repeated emphasis on the solubility of CsI and other "metallic iodides" (Stratton, p. 19) as opposed to a presumed "insolubility" of I_2 ascribed to the RSS model. The authors of the Stratton testimony apparently interpreted the RSS methodology to ascribe "near-noble gas" behavior to I_2 (Stratton, p. 21). In reality, the RSS methodology ascribes very high solubility to I_2. For an accident sequence of the TMI type (i.e., the release from the core contacts water prior to release to the environment, and pH adjusted containment sprays are operational) the RSS's CORRAL code calculates that the iodine concentration in the liquid phase is more than 10,000 times higher than that in the containment atmosphere (WASH-1400, App. VII). Therefore, it is necessary to perform a careful examination of the dynamic behavior of radionuclide transport mechanisms based on the expected thermal hydraulic conditions in the primary system and containment in order to assess the effects of the chemical form of the iodine.

3. Retention in the Primary System

The Stratton testimony correctly points out that the RSS analyses of fission product transport do not account for agglomeration and deposition of fission products in aerosol form within the primary system. We agree that this is a known conservatism of the RSS and other analyses employing the RSS methodology. The Stratton testimony discusses the potentially important phenomena in the reactor vessel (Stratton, p. 31ff). However, the Stratton testimony quotes from a 1977 publication which describes the processes important in a "terminated LOCA," i.e., a design basis accident for which no substantial fuel degradation or melting is assumed (i.e., the TRAP-LOCA code). We believe that a more appropriate basis for estimates of primary system retention is the next level of development of the TRAP model, i.e., the TRAP-MELT code which addresses core melt conditions (NUREG/CR-0632, 1979). Calculations of primary system retention during core-melt accidents using the latest version of this code are currently in progress. Although it is premature to make generic conclusions on the basis of these calculations, the results (as stated in NUREG-0772) indicate that the degree of radionuclide retention in the primary system is highly dependent on the specific accident sequence and the specific reactor design, and suggest that categoric "reduction factors" cannot be supported at the present time.

4. Fission Product Behavior in Containment

In order to characterize the behavior of fission products in the containment, the Stratton testimony refers back to the Containment System Experiment (CSE) of the late 1960s (Stratton, p. 40ff) to demonstrate the effectiveness of containment spray in removal of elemental iodine and aerosol particles. The authors of the Stratton testimony reject the quantitative predictions of radionuclide behavior in the containment provided by the RSS's CORRAL code. However, it should be noted that the CORRAL code is based on an empirical fit to the CSE data. Better predictions of radionuclide behavior in containment are now being validated, such as the NAUA-4 code, but the staff feels that it is still somewhat premature to base source term calculations in a proceeding such as this upon them.

The above examples illustrate the staff's difficulty with the method used and some of the conclusions reached by Stratton, Rodger and Potter. We believe that quantification of "reduction factors" to be applied to source term estimates is premature.

The NRC research program is actively engaged at this time in developing substantial new information regarding severe accident source terms for light water reactors. The approach by which this is being done is rather complex; it involves the development and application of new computer codes to describe the important processes in the core region, the reactor coolant system and the containment during the degradation and melting of the core. In addition to code development, extensive experimental work is in progress here in the United States, and in foreign countries as well, to augment the data base for the verification and validation of these codes.

Using the codes developed for this purpose, a series of U.S. light water reactors are being studied, one at a time, taking significantly different accident sequences and carefully analyzing the release and transport of radioactivity during those sequences. For example, the first plant studied in this manner, the Surry PWR in Virginia, is being studied for the accident sequences large break loss-of-coolant-accident, small break loss-of-coolant-accident, station blackout, and what is called in the Reactor Safety Study, Event V, the interfacing system LOCA where the reactor coolant system ruptures directly outside the reactor containment building. After these selected sequences are analyzed, a careful appraisal of the dominant accident sequences for a plant of this type can be conducted to appraise the expected accident source terms for all of the dominant accident sequences, and thereby develop the compendium of source terms which describes the risk characteristics of that plant. In addition, a report describing the technical base for these computer analyses is being developed on a parallel schedule. In addition to the NRC-sponsored work, we expect to see published this coming summer the results of similar work by the industry degraded core group (IDCOR) as well as an interim report on the subject from an American Nuclear Society special committee chaired by Dr. Stratton.

Later this year, after it has undergone peer review, the staff of the NRC will appraise this technical data to determine what substantive changes in accident source terms may be justified at this time and will then advise the Commission of the significance of these results as well as proposing to the Commission what regulatory or policy changes might be appropriate based on these accident source term changes. Therefore, we believe that at this time it is premature to attempt a quantitative reassessment or restatement of accident source terms here in the Indian Point proceeding or in other cases as well.

STATEMENT OF QUALIFICATIONS
ROBERT M. BERNERO

I am Robert M. Bernero, Director of the Accident Source Term Program Office in the Office of Nuclear Regulatory Research at the U.S. Nuclear Regulatory Commission.

I hold the degrees of Bachelor of Arts (1952) from St. Mary of the Lake (Mundelein, Illinois), Bachelor of Chemical Engineering (1959) from the University of Illinois, and Master of Chemical Engineering (1961) from Rensselaer Polytechnic Institute.

Early in my technical career I was employed by the General Electric Company at the Knolls Atomic Power Laboratory from 1959 to 1966 where I worked on the design, construction, and test of pressurized water reactors for naval propulsion plants. In that work I gained substantial experience in reactor electrical and fluid systems as well as the chemistry and radiochemistry of reactor cooling systems.

From 1966 to 1972, while still with the General Electric Company, I worked at the Valley Forge Space Center where I participated in the development of nuclear power devices for space applications. In my final position there I served as Manager of Energy Conversion Engineering where I directed the design and development of a high temperature thermoelectric power system using silicon-germanium alloys.

In 1972 I joined the Atomic Energy Commission regulatory staff as a licensing project manager. From 1972 until 1975 I managed reactor licensing cases including pressurized water reactors and the Clinch River Breeder Reactor. In 1974, when the draft version of the Reactor Safety Study (WASH-1400) was published, I was a member of the NRC staff team which performed an in-depth review of that benchmark risk assessment.

From 1975 until 1977 I worked in the NRC's Office of Nuclear Materials Safety and Safeguards as the licensing manager of the Barnwell Nuclear Fuel Plant and as Chief of the Fuel Reprocessing and Recycle Branch.

From 1977 to 1979 I served in NRC's Office of Standards Development as Assistant Director for Material Safety Standards. After the Three Mile Island accident, I served in the staff of the NRC's TMI Special Inquiry. At the end of 1979 I was appointed to be the Director of the Probabilistic Analysis Staff in NRC's Office of Research. That group, now known as the Division of Risk Analysis, has long been the center for the development of risk analysis methods at the NRC.

My permanent position at the NRC is still Director, Division of Risk Analysis. However, in January of 1983 I was appointed to be Director of the Accident Source Term Program Office. This program office was formed in January to assure that source term related research results are implemented in policy and regulatory practice in a timely manner.

MS. MOORE: Your Honor, the witness is now available for cross-examination.

MR. BLUM: Your Honor, given the positions of the parties on this testimony, we would request the Licensees go first with their cross-examinations since they are in the adversarial stance.

JUDGE GLEASON: All right. That is a point well taken. Who wants to proceed?

CROSS-EXAMINATION

BY MR. COLARULLI:

Q. Good morning, Mr Bernero.

A. Good morning.

Q. You are familiar, are you not, with Drs. Stratton and Rodger, and that this data available today strongly suggests that the WASH-1400 models overestimate the amount of radionuclides released in the environment?

A. Yes, I agree that the data suggests that to be true.

Q. I wonder if you could turn to page 3 of your testimony.

A. I have it.

Q. You refer to several physical processes which would affect some degree of retention or rate of nuclides in various parts of a facility as being neglected in their reactor safety study methodology.

A. Yes.

Q. Specifically, what are those processes that you believe are neglected?

A. Examples of those processes are, one, the process by which chemicals such as iodine, rather than behaving in its elemental form, would combine with another chemical and become a compound with different physical characteristics.

A second example would be the formation of large amounts of inert aerosols, that is particles that are not of direct interest as radionuclides, which could aglomerate, come together, and absorb radioactive aerosols.

Those are just two examples of the sort of processes I'm thinking of.

Q. Are these processes, the formation of cesium iodide and the aglomeration that you referred to, are these processes addressed by Dr. Stratton and Rodger?

A. Yes, I believe so.

Q. Do you agree with their descriptions of these processes?

A. In general, yes, their qualitative descriptions of the potential working of these processes.

Q. Do you consider Drs. Stratton and Rodger to be reputable, competent scientists?

A. Yes, I do.

Q. What is Staff's view with regard to the impact of iodine, in fact, forming in the cesium iodide? What impact does that have on the source terms and ultimately on health effects?

A. Well, in general, the formation of cesium iodide in or near the core will probably result in the iodine being less mobile, less able to get out of the reactor coolant system in a coremelt accident, and less mobile once it's in the containment building after leaving the reactor coolant system, so that in any given accident sequence, less iodine might be released.

Now, the radioiodine, by present estimates, constitutes approximately half of the early fatality risks, so it could have a significant effect on offsite health effects, which are calculated.

Q. I'd like to turn to table A-1. Do you have a copy of Drs. Stratton and Rodger's testimony before you?

A. Yes, I do.

Q. And could you turn to table A-1?

A. I have it now.

A. Table A-1 is entitled "Evaluation Of Environmental Releases For Indian Point Sequence 2LW."

I'd like you to look at the first entry, Time: Hours, zero," and reading across those items, the "Event Sequence," the "Expected Behavior Of Cesium Iodide," then the "Fraction Of Core Inventory."

Looking at those entries, do you have data or information that suggests that these descriptions and statements are wrong?

A. No, I do not.

Q. Looking at the next time period from two hours to 2.5 hours, and, again, at the "Event Sequence" and the "Expected Behavior" and also at the "Fraction Of Core Inventory," do you have any data or information that suggests that the values under the fraction of core inventory, both in containment air and also released to the environment, are wrong?

A. No, I do not, but I would make a

general comment here, that starting with this step in the table and going on through all the other steps, I feel we are dealing with the exercise of scientific judgment, a judgmental analysis, which says that a certain fraction of the radionuclides will go one way and a certain fraction go the other way.

As we, and I mean myself and the Staff at the NRC, as we go through this table, we don't have or find data to contradict what we see here, nor do we find rigorous data or modeling, validated models, to confirm what we have here.

We believe it's a judgmental analysis and considering the work that's underway. We consider it, as I said in my testimony, premature to quantify it as rigorously as this.

Q. Is it not the case that even following the additional research that you reference in your testimony, that even with that research, there will still have to be judgments made since there's not been the kind of accident that we are assuming to model and quantify in terms of the chemicals released?

Isn't there going to be scientific judgment that you are going to exercise at the end of your evaluation?

A. Oh, yes, there's certainly scientific judgment to be exercised at the end, but there will have been far more direct physical experimentation to give us direct knowledge of how some processes work under degraded fuel conditions, and that same data may be extremely useful in validating predicted models for such behavior.

Q. You take issue with several points in the Stratton and Rodger testimony, I believe beginning at page 5 of your testimony.

A. Yes, that's correct.

Q. How would you characterize your position with regard to these particular items?

Would it be fair to say that you consider Drs. Stratton and Rodger's analysis fundamentally wrong, or are these rather some misinterpretations that you believe exist in their analysis?

A. I would characterize them as misinterpretations that I see in their analysis, the TMI experience emphasizing, as an example, the amount of radioiodine that was released.

As I said in my testimony, I don't think that that is the proper comparison or useful comparison for TMI, because it was not a full-fledged coremelt and did not involve severe failure of the containment.

There was a prolonged bypass leakage through a fluid system, the letdown and makeup system, which does not give a direct comparison to WASH-1400.

TMI, as I said in my testimony, is useful, nevertheless, for the evaluation of what goes on or went on inside the core and inside the containment.

Q. Do you recall the deposition of March 31 that the Licensees took of you?

A. Yes, I do.

Q. And do you have a copy of that deposition in front of you?

A. Yes, I do.

Q. Could you turn to page 76 and 77.

A. Yes, I have it here.

Q. At the bottom of that page, in response to a question concerning these differences that we were just referring to, you state, "A number of people in the Staff reviewed it and did not find criticisms except in a few places that we single out in the testimony where we thought there was some misinterpretation of the WASH-1400 model as such, but they are subtleties more than anything else but in the tables and the scenarios given that these are the dominant accident source terms and the IPPSS as we look at them. We can't say whether they are right or wrong. They appear to be reasonable."

Is that still your opinion?

A. Yes, that is still my opinion. When I say "right or wrong," I mean right or wrong quantitatively.

Q. What is your personal expectation with regard to the direction of the change to the radionuclide release terms that are referred to in the Stratton and Rodger testimony? By what factor do you anticipate change?

A. As I said in my testimony and as I have said before, the oversimplification, we are all guilty of at times, is to speak of a single source term and a single reduction factor.

I would say, again, we should be on guard that the source term is a set, a spectrum of source terms.

I have said before, and I feel as a personal prognosis, that there is a reduction in overall releases on the order of a factor of ten. That's just a personal belief. It is not supported by rigorous analysis yet in anyway that I know of, and, therefore, I don't think it's trustworthy for regulatory use at this time. It would be premature to use it.

Q. However, if your personal prognosis is accurate, would you consider that to be a significant reduction?

A. Yes, indeed, and having made that personal prognosis along with other knowledgeable members of Staff, we have taken pains to do sensitivity analyses from time to time to explore what effect such changes in source term magnitude would have on the calculated consequences of reactor accidents, and they are quite significant.

Q. Do you recall generally what kind of impact of reduction a factor ten would have on, for example, early fatality risks and late fatality risks?

A. Well, I can only point to a sensitivity analysis in the Sandia siting report which was published under our sponsorship last November, and I don't remember the exact values in there, but there is a sensitivity analysis for various reductions in source term, and it has the most dramatic effect on early fatality risks and less dramatic effects on injury and latent cancer and property damage risks, but I don't recall the nuclear values.

Q. Is it your opinion that the key radionuclides, in terms of health risks, would include cesium and radioiodine and tellurium?

A. Those are among the most significant radionuclides in the core for health risk purposes.

Q. In given a late overpressurization scenario, how differently would those radionuclides behave at, say, eight hours into that kind of a scenario as compared to an early overpressurization failure?

A. This behavior was described in some detail in a document that was published by the NRC in 1981, called NUREG-0772, a very long title called, "The Technical Basis For Estimating Fission Product Behavior In LWR Accidents," and in that document, there is a curve which displays this as modeled by a variety of computer codes which are now available.

For delayed containment failure, physical processes that work in the containment building, are working on the suspended radionuclides or aerosols and dropping them down so that they won't come out once the containment fails or are far less likely to come out once the containment fails.

Slow overpressure failure is such with these physical forces that the longer you delay, the more you reduce the source term and subsequently the offsite consequences, and the effects can be quite dramatic. I can't give you the nuclear values from that curve, but they can certainly be looked up.

Q. When you say "dramatic," is that synonymous with significant, which is to say --

A. Well, yes.

Q. Would you anticipate a factor of ten reduction after, say, eight hours, or can you give us any kind of general parameters on that?

A. Yes. In general, one can achieve, if you look at that curve by substantial delay in containment failure time, you can achieve significant or dramatic changes in source term, and by that, I mean a factor of ten or more.

Q. You just referenced NUREG-0772. Is it your view that NUREG, in discussing radionuclide release terms, that, in effect, NUREG finds that the uncertainty band has shifted dramatically and that, therefore, is much more likely that the radionuclide release terms are lower than in WASH-1400?

A. I'm not sure that NUREG-0772 contains within it the words "shifting the uncertainty bands," such as that.

It is a way I choose to describe it. The document concludes that that releases from coremelt accidents are likely to be substantially lower than as presented or analyzed by the Reactor Safety Study Models, and that document goes on with some discussion of what ways it might be lower, and I think a fair characterization of it is that it describes the Reactor Safety Study Model as still being a usable point estimate, but the uncertainty about it is now shifted so there is very little likelihood that the releases would be higher, and a much greater likelihood that the releases would be lower than characterized by that model.

Q. NUREG-0772 was published sometime in 1981; is that correct?

A. That's right.

Q. During the past two years since publication of that NUREG, do you say that the additional information and data that is now available since that time would also suggest that the releases would be more likely lower than in WASH-1400, assuming that the mechanisms, the processes that you referred to earlier are, in fact, at work?

A. It still is a little too early to say what the 1983 reassessment tells us. We have conducted only a modest portion of it, which went into the peer process leading toward the ultimate publication of a sequel to NUREG-0772, and we have had the peer review of that first portion.

I feel that the corrective actions we are taking to respond or in response to that peer review are so complicated that it's premature to say whether our current reassessment in 1983 will be stronger or less strong than NUREG-0772 was two years ago.

Q. Well, let's try to be more specific, then.

Regarding the behavior of tellurium, for example, what indications, what conclusions can you draw today based upon the data and the information available as compared to those drawn in WASH-1400?

A. Well, we can see today and have been discussing vigorously in the process of doing this work the possible mechanism by which tellurium, a relatively volatile and one of the more noxious radionuclides mechanisms by which tellurium can react with other metals, forming alloys, which could get it out of the process, get it out of the source term to a substantial degree.

It could release from the core and find its way to the stainless steels in the upper plenum forming an alloy of sorts and thereby be removed.

On the other hand, there are some scenarios, some mechanisms, whereby if a great deal of the zirconium is still available unreacted, that tellurium might react first with the zirconium and in some respects get greater mobility to get out of the reactor coolant system. It's a complex process.

Q. Assuming that the first process you described in which the tellurium in effect attaches itself to other surfaces, what effect would that have in terms of lowering the amount of tellurium that's available to get out? Is it small? Is it dramatic? Can you put any kind of a quantification on that given the research and the data available for the past seven years?

A. Well, if I take your assumption that the tellurium got up, say, into the upper plenum and reacted with the stainless steels up there, then it would be far less likely to be remobilized and made available for the source term later on in the accident. It could reduce the source term.

Q. And can you say, now, is it too early or can you say now whether or not that might be significant?

A. I can only say it could be.

Q. Could be significant? Could be significant on the order of other things that you have characterized as in the reduction of a factor of ten?

A. As I said before, it depends on the specific analysis. The tellurium could get out by other mechanisms, and depends on what these competing forces result in, just how much one should properly estimate as the release of tellurium from the reactor coolant system and in what form it would be released.

Q. We have, as you know, and as you have heard earlier today, had discussion of mitigation devices.

I'd like to ask you this question in that regard: Do you believe that it is appropriate to make decisions regarding whether to add certain mitigation devices, such as a filter vent, using what you would agree would be an immediately unrealistically high radionuclide release term as in WASH-1400?

A. I think that regulatory decisions, such as whether to add a mitigation system to a plant to improve its safety can and must be made with estimates that are known to be biased or skewed in some way.

The source term, which everyone chooses to call the WASH-1400 source term, which is the current estimate for radioactive releases in accidents, is a fair basis of judgment if the persons making the judgment have a reasonable understanding of the biases and uncertainties in that source term estimate.

Q. But do you not think that it would be more appropriate to, before making decision on a substantial mitigation feature, to have evaluated all these processes and available technical data to which you refer and to come up with what the community could agree upon as being a realistic source term instead of an unrealistically high one?

A. In an ideal world, yes. That would be -- the nice way to do it, we have at the NRC an extensive program that is reasonably well coordinated with work by the Electric Power Research Institute and work in foreign countries, and we expect to see the nickest results or the most clear results in 1985.

If one is patient and able to wait until 1985, it would be a very nice time to make such a judgment.

In 1981, we felt the need to speak what we knew then, and that was NUREG-0772, and in 1983, we are preparing a document which will speak what we know today, in 1983, recognizing that many proceedings, many needs carry on and may not be able to wait until 1985.

Q. But if the source term, the ultimate one that is agreed upon at a given time is, in fact, significantly lower than what we all call WASH-1400 source terms, is the value of the mitigating device also significantly lower?

A. Generally, I would say it would be. If your depiction of risk is lower to begin with, then any device whose purpose is to further lower risk has less to work on.

MR. COLARULLI: I have no further questions.

JUDGE GLEASON: Mr. Brandenburg?

CROSS-EXAMINATION

BY MR. BRANDENBURG:

Q. Mr. Bernero, in the most recent line of questioning from Mr. Colarulli, I understood you to say that some understanding of the biases of the present source terms would be necessary in order to make and form judgments from those source terms for purposes of deciding on mitigation strategies.

I'd like to ask you a slightly broader question in which we might have to lay to one side your assumption about an understanding of these biases. I'd like to just ask you as a general matter whether you agree that overly considerate source term release estimates lead to inaccurate and misleading overestimates of the consequences of various accidents, again, apart from this understanding of the biases that you have assumed.

A. If I understood you correctly, your question, I think it answered itself, that overestimates lead to understanding of greater risk than really is there, something to that effect.

Q. All right. Absent this understanding of the implicit biases that you referred to, would overly conservative estimates of the magnitude of the source terms comprise an inappropriate basis for regulatory judgments? In other words does this component of understanding the biases, is that essential in your judgment to reaching appropriate regulatory judgments?

A. Yes, it is. That understanding of the bias, or, to use another term, the uncertainty in estimates, is necessary to reach a responsible judgment.

Q. Now, I believe you were seated here when I questioned Dr. Meyer and Mr. Pratt about the interrelationship between source terms and the analysis of mitigation features.

Would you agree that one should estimate the source terms correctly in order to accurately estimate the risk reduction effectiveness of mitigation features?

A. If one is careful in understanding the word "correctly," yes, one should use the appropriate current estimate of source term, recognizing its uncertainty, conduct an evaluation of a mitigation feature and keeping that uncertainty in mind, compare the risk reduction of such a mitigation feature to whatever figure of merit is available.

Of course, one recognizing the figure of merit does not come down on a bronze tablet, so that the uncertainty judgment pevades the whole process.

Q. Now, the correlation between the source term and the computed benefits from mitigation devices is a positive one, is it not, Mr. Bernero, in the sense that to the extent that the source terms are overstated, one's analysis using those source terms of the effectiveness of the mitigation device will similarly be overstated; is that correct?

A. Generally, as I said before, that's true. If the source term is overstated, the accident risk is overstated, and, therefore, if there is less risk to work on, there is less effectiveness for any mitigation feature to work on.

Q. Now, as I understand the assumptions that were made in WASH-1400 and implicit in it those source terms which you indicated are currently being used by the NRC with respect to the behavior of cesium, and I'm actually thinking back to a passage in your deposition from last week that Mr. Colarulli referred to, and I think you have a copy before you.

I understood that your general perspective on that was that the investigators in WASH-1400 did believe that cesium iodide was probably the preferred chemical form for iodine to be in, but that there was a limited amount of data but that might not be so. So to be conservative, they treated iodine as elemental.

Is that a fair summary?

A. Would you please cite the page number of my deposition.

Q. Surely. I'd be happy to. I believe the discussion starts on page 18.

A. Yes, I recall.

Q. Do you recall the quotation at line nine, I think?

A. Yes. In Appendix 7 of WASH-1400, which is the physical processes appendix, there is a discussion, as I recall, of the free energies or thermal dynamics of chemical reaction which indicated to those investigators that cesium and iodine were likely to combine chemically in or near the core, especially since cesium outnumbers iodine by a very large margin.

However, they cited some data, and I don't remember the researcher or citation, but I'm sure you could find it in there, that suggested that it would not form cesium iodide under those circumstances in core degradation.

And so they assumed that for analytical purposes, cesium and iodine would not combine, and they would be, therefore, volatile materials and come almost directly out of the core into the containment building.

Q. Now, with specific reference to that limited amount of data, I think you referred to it, as that suggested that cesium iodine would not be produced, that led to the Wash-1400 investigator's decision to use an assumption of elemental iodine.

Could you describe for us your understanding, if you have one, as to the rigorousness or the quality of that data?

A. Oh, no, I couldn't. I'm not familiar with that data, and I know of it only historically as a matter that was presented in the discussion in that appendix of WASH-1400 with some of the early data in this field. That's all I know.

Q. Now, we are moving forward in time to the work that's going on now by -- under the direction of your division on this -- is it cesium iodine or is it the elemental iodine issue?

Can you for us, describe for us, how that experimental activity is ongoing?

A. Well, there's a good deal of experimentation going on and has gone on in the ten years or so since the Reactor Safety Study Analysis was done, and in NUREG-0772, two years ago, it was expressed as a, I think the term was technical consensus. I'm not sure of the words -- that the expected form in this circumstance was cesium iodide, not elemental iodine, and that is based on a variety of experimentation of, oh, what you might call fuel roasting experiments where spent fuel is heated and careful measurements are made of what comes off and what condenses and where it does.

Q. Now, has there been anything, any experimental activity that's gone on subsequent to NUREG-0772 that has tarnished that scientific consensus that cesium iodine would be the preferred form of iodine?

A. Not that I know of.

Q. Now, you covered the topic of tellurium with Mr. Colarulli, and I would like to finish out those isotopes that would contribute to the early fatality risk, and now I'm really asking you, Mr. Bernero, to distinguish in your mind between those isotopes that contribute to latent fatality risk on the one hand and early fatality risk -- concentrating only on the early fatality risk isotopes -- and ask you soley with respect to those isotopes making a significant contribution to early fatalities, if we have failed to cover any.

Is tellurium iodine pretty much it or are there others in here?

A. Well, others get in. I recall that ruthenium, there is a one year half life ruthenium that can play a reasonably substantial role in early fatalities. There are a few other nuclides.

It's getting a bit beyond my direct expertise to go into the individual nuclides and their contributions.

Q. Now, I think you stated in response to a question from Mr. Colarulli that while the Staff has a good deal of ongoing experimentation in this area, that your own personal expectation -- I think you did hedge it to that extent -- was that a reduction of a factor of ten seemed plausible to you.

Is that a fair statement of your personal expectation?

A. Yes. That, I think, is the phrase that I used, either "personal expectation" or "personal prognosis". One does this regularly in doing research work to set some priority, where should we look, how hard should we look, what resourses should we dedicate, and it guides your decision process in some way; and that is, again, a personal expectation, but not proven.

Q. Of course. Dr. Bernero, I would like to follow through with one more step with your personal expectation, and I'd like you to do that with me, if you could, by turning to page 64 a of Dr. Stratton and Dr. Rodger's testimony.

A. Excuse me. It's Mr. Bernero.

Q. I also mischaracterized our own testimony. It's the testimony of Stratton and Rodger and Potter.

A. Yes, 64-A is a figure?

Q. Yes, sir, it's figure 2.

A. Yes, I have it before me.

Q. Now, there are two curves shown there, and I would like to focus your attention on the lower two curves, which deposits a factor of ten reduction from the source terms that were used in the IPPSS study, and my question is, if your personal expectation about source terms turns out to bear fruit at the end of the Staff's current exploration of this topic, and if you further accept the consequence model as it was done here, that a diffusion analysis and the evacuation analysis and that sort of thing, that based upon the analysis of these gentlemen and assuming that your reduction of a factor of ten, as I say, does bear fruit, that, in effect, we would have with a frequency of less than ten to the minus eighth and no more than about 20 fatalities from a serious accident at Indian Point?

A. I think it's time to register the

comment about oversimplification when one speaks of an overall reduction in source terms by a factor of ten then going into an individual reactor profile.

I think you should stop at the qualitative description. I would agree that in a plant such as Indian Point, that qualitatively, a substantial decrease in early fatalities, et cetera, would be expected, but whether that particular probability or that particular level of consequences, I can't testify to that.

Q. All right. I understand that you have a difficulty with accepting the assumption of a factor of ten reduction now certainly for this sequence, but you have reviewed, of course, the Stratton and Rodger and Potter testimony; have you not?

A. Yes, I have.

Q. And have you performed any analysis of the rest of the modeling that was done leading up to this figure after the assumption about the reduction and source terms? Have you discussed it with any of the consequence people at the Staff, for example, once you take the initial assumption of the factor of ten reduction, whether there are any problems arriving at this curve shown on figure 2 that we were discussing?

A. In our review of the testimony, we made no attempt to go into the individual reductions calculated as such. Our attention focused on the source term reduction factors and the method by which they were obtained. To

go beyond that point, I feel, was presumptious, that the reduction factors were more rigorous than we thought them, and, therefore, we would have further review to do.

We did not separately analyze these figures to see whether we would have gotten the same results.

Q. But after one gets by the initial assumption, I understand that you have some difficulty with it at this point in time, the initial assumption of a factor of ten reduction, the Staff is unaware of any other modeling errors, things of that nature, leading up to this preparation of this figure; is that right?

A. I just can't comment on it. I don't know.

Q. Now, are you familiar, Mr. Bernero, with a document that's been widely referred to as the Sandia Siting Study?

A. Yes, indeed.

Q. And I think we can identify that as NUREG-2239?

A. Yes. In my normal position at the NRC, my division there, the Division of Risk Analysis sponsored that work.

Q. Now, was it a conclusion of that document, the Sandia Siting Study NUREG-2239 that the estimates of the early fatalities from serious accidents were very sensitive to the magnitude of the source terms?

A. Yes, indeed. I don't recall the section number, but there is a section in that report which conducts a sensitivity analysis to those simplified source terms, and they were explained in that report and supporting documents. A sensitivity analysis was done to explore what differences in offsite consequences would be derived from different reductions in source term, and there's a whole section devoted to that in that report.

Q. Now, can you describe for us, Mr. Bernero, in a general way, the relationship between source term reduction and early fatality consequences that was found by the Sandia Siting Study? How much reduction in early fatalities do you get for how much reduction in source term, generally?

Obviously you don't have to answer this in a general way.

A. I don't remember the exact numbers, but there were tables presented there which showed that with across-the-board reductions of radionuclide release.

In other words, across-the-board reductions of source term, that the early fatalities would drop most rapidly, early fatality being a threshold effect. You have to recieve a certain dose, fairly high dose, before there's a threat of early fatality.

The second most rapid reduction would be in early injuries, radiation injuries which is also a threshold effect, although not as grievous a radiation dose as early fatality dose.

And, lastly, there was a much lower sensitivity, although still, as I recall, almost linear sensitivity in the latent fatality and property damage effects, which are dominated by nuclides like cesium 137, things with a long half life.

Q. Now, again, focusing only on the early fatality situation and leaving aside the latent injuries and so forth, do you have any general understanding what the conclusion was on the Sandia Siting Study as to how much the across-the-board source terms would need to be reduced in order to have no or virtually no early fatalities?

A. I don't remember the numbers in it. They had tables that brought it down, as I recall, two orders of magnitude or more, a factor of 100 or a factor of 1,000 but I don't remember the nuclear results.

That was only a secondary thing. That result was focused on siting parameters and the risk significance of siting parameters in that case, and that sensitivity analysis was to provide the regulatory thinker with an appropriate understanding of the significance of the uncertainty in source term.

Q. Now, again, Mr. Bernero, I was struck by a passage in your deposition from last week in which you said, generally, that "A reduction in source terms for iodine were accident sequence specific and plant geometry specific."

I can give you a page, but perhaps you just generally recall that.

A. Could you give me the page, please.

Q. Sure. It's page 23.

A. Yes.

Q. Top of page 24, actually.

A. Yes, I recall the passage now.

Q. Given that opinion of yours, what uncertainties, in your judgment, are introduced by employing the WASH-1400 source terms that model the Surry Palnt and applying those source terms in an effort to model the releases that one would expect to find from the Indian Point Plants?

A. Well, I believe that on comparison, if you looked at the Surry Plant and the Indian Point Plant, you would find that the containment in the Indian Point Plant is in -- in either one of the Indian Point Plants, is stronger, larger, than the Surry Plant.

Also, you would have different probability distributions for the different accident sequences, that is the relative likelihood of blackout as against loss of coolant accident or some other accident sequence.

One of our concerns about the limited surrogate character of the Surry Plant lead us to add the analysis of a fifth plant to this body of work, the Zion Plant, which is fairly close to Indian Point, I believe.

Q. You mentioned the containment strength. What other plant specific features would be important to an evaluation of source term behavior?

A. The distribution of accident sequences, the relative likelihood of one sequence over the other, because each sequence has its own source term characteristics, different timing of the core deterioration and melting, different path ways for the noxious radionuclides to travel along,.

Q. Now, in your answer contrasting Indian Point with Surry, I think you have said that the Indian Point containments were larger, but what effect would you expect that to have on source term behavior?

A. There is a risk significant question in any of these reactor analyses, and that is to evaluate or assess the vulnerability of the plant to early containment failure. The reason for the interest is that the best known phenomena which stand to reduce source term are those in the containment atmosphere, and if the containment fails early in the accident, then those phenomena, those forces in the containment, have little or no time to work on the source term.

Q. So if I understand you correctly, the longer the time in which the -- the longer the holdup time of the materials inside the containment, these forces and processes that you referred to would result in a reduced release when the release actually were to occur; is that right?

A. That's right. The larger, stronger containment, all else being equal, will hold the material longer and permit further time for material to aglomerate and plateout on surfaces or fall to the water at the floor and be out of the source term.

Q. Now, to your knowledge, in the connection with the Staff's testimony for this proceeding, did they attempt to model even in any sort of gross way the differences in source term because of the plateout and the other things that you have just referred to that might be expected at Indian Point versus what might be expected at Surry and more implicit in the source terms used in WASH-1400?

I guess we are talking pretty much pressure of time and things of that sort.

A. I can't testify as to how they treated the containment failure time, and I can only note that I heard Dr. Meyer or his colleagues say this morning that they used CORRAL 2, which is a refinement of WASH-1400's model for precipitation of aerosols, but I can't testify as to how they treated the pressure or failure time.

Q. Now, Mr. Bernero, in the course of your deposition, you caught my attention with a reference to something called a mosquito curve.

Could you tell us what that is and how that affects the source term behavior that we are talking about?

A. We have fallen to the use of that term within the NRC because of the resemblance, physical resemblance, of all the lines on that graph.

If you visualize a mosquito sitting on your arm facing to your left as you look at it, there are a series of curves like the feelers and the proboscis and the front legs of the mosquito all coming up together giving essentially the same prediction of aerosol concentration versus time coming to some peak, as the core degrades, and, then, the rear legs of the mosquito extending out on a very shallow angle to the rear of the insect.

The upper unsupporting legs which are falling at a much slower rate or inclined in a much more shallow angle than the other legs, describe the WASH-1400 CORRAL Code, and the way it models the fallout of aerosols in the containment.

Other codes, such as the NAUA Code, fall or predict fallout of aerosols at a more rapid rate, and, therefore, time that is delay in containment failure is even more effective of source term reducer.

Q. Now taking a factor of ten reduction for iodines as sort of a yard stick and looking at these mosquito curves that you just referred to, after what period of holdup time in the containment does one obtain a factor of ten reduction in iodine?

A. I don't remember the exact numbers from that curve. It's on the order of hours, eight hours or ten hours or something like that.

Q. Now, you mentioned the use of the NAUA Code. Can you tell us what role that displayed in the Staff's ongoing examination of these issues?

A. The NAUA Code or the NAUA 4 is the version that we consider for use now. It is being used instead of the CORRAL Code in our present estimates of aerosol behavior in the containment.

At this time, it appears to be the most valid model available for such prediction.

Q. Now, when one uses the NAUA 4 Code versus its analogue in the WASH-1400 analysis; and when one looks at the late containment failure accident sequences, does not the NAUA 4 Code result in approximately a factor of ten reduction in the source terms?

A. At some delay time that is containment failure time, yes. Those curves I described in NUREG-0772, you can read the difference at any given containment failure time. The longer the containment holds, the more dramatic is the reduction you calculate with NAUA as compared to CORRAL, which is the WASH-1400.

Q. Now, while we are on the subject of codes, Mr. Bernero, you are aware, I believe, that the Staff used the March 1 Code in its analysis of the IPPSS study as to the IPPSS investigators?

A. Yes, I'm aware of that, the March 1.1 Code or some derivative of it.

Q. And is it your view that when one employs the March 1.1 Code, one models a more rapid core heatup than you would expect?

A. Overall, yes, that appears to be the case. We are just now comparing the results of the improved March Code, which is called March 2.0, and after incorporating all of the changes, it appears that the March 2.0 code gives an overall slower coremelt time prediction, that is the heatup to coremelt, it takes longer, although the individual shape of the curve can be different.

Q. Now, Mr. Bernero, specifically with reference to the Stratton and Rodger and Potter source term testimony, Mr. Colarulli showed you a few entries on table 1.A, and you testified that you had no information or data suggesting that the behavior would be otherwise.

As a general matter, based on the Staff's current examination of source term issues, is there information or data to suggest that source term behavior would be otherwise than as discussed in the Stratton and Rodger and Potter testimony?

A. I presume you are referring to table A-1 in the -- as the first of that series of tables in the Stratton-Rodger testimony.

Q. Yes.

A. No. The Staff doesn't have data that suggests contradiction of this walkthrough and quantification of the source term, so much as the Staff sees it as a more complex analysis being needed, the complex interplay of heat transfer and mass transfer, as you go through the accident scenario, which is the substance of the work we are doing today.

Q. So would it be a fair statement in your view that the Stratton-Rodger-Potter reductions may be demonstrable? You just don't know at the present time?

A. We just don't know yet.

JUDGE GLEASON: Exhausted this subject, Mr. Brandenburg?

MR. BRANDENBURG: Why, Mr. Chairman, you must be reading my mind. I think that was my last question.

JUDGE GLEASON: Thank you. We'll stand at recess, Mr. Blum, until 2:00.

(Hearing adjourned at 1:00 p.m.)

(Hearing reconvened at 2:10 p.m.)

JUDGE GLEASON: All right, if we could proceed, please.

JUDGE GLEASON: Mr. Blum?

CROSS-EXAMINATION

BY MR. BLUM:

Q. Mr. Bernero, you recall talking about the rough estimate of a factor of ten reduction in source term; is that correct?

A. Yes. When I say estimate or you say estimate here, I believe you refer to what I call an expectation or a personal prognosis, and, yes, I do recall talking about it.

Q. Right, and you testified that that's very approximate in two ways; one that it is only a prognosis, and the other is that it lumps together a large number of separate figures which might be quite different from one another; is that correct?

A. Yes, indeed, that's so. The number of separate figures would cover the spectrum both of different radionuclides and chemical forms thereof as well as different accident sequences where the behavior can change.

Q. So just to make things very

unequivocal, you are not suggesting that the Board or anyone else should rely on a factor of ten source term reduction from WASH-1400, are you?

A. No, not at all.

Q. You are familiar with the peer review meeting that occurred on January 25 and 26, 1983 for the NUREG-0956 draft study of accident source term?

A. Yes, I am familiar with that proceeding. I did not attend, and I would like to insert a clarification. That was the peer review discussion of a portion of NUREG-0956. It was the analysis of but one plant, and it will be a relatively modest portion of the whoel NUREG-0956 document.

Q. Are you familiar with the attenuation factors for iodine and for cesium that were being discussed as rough working hypotheses in that meeting?

A. To some extent. I did not attend the meeting, but I have read all of the peer comments that derive from the meeting and have had Staff reports and clarifications that might be appropriate based on the meeting.

So I'm not sure what you are

referring to when you say "the attenuation factors."

Q. Well, the specific thing to which I was referring were table 7.14 and 7.15. The specifics of which -- I don't know if we have -- go ahead.

A. If I could interrupt you, I think I understand what you are referring to now. You are referring to the calculated release or source term fractions that were in that report, and there was a good deal of discussion comparing them to the reactor safety study equivalent fractions. Is this the --

Q. Yes.

A. Those tables you cite sound like the ones that do that.

Q. Yes, that's correct, and you do recall generally that there were a number of release fractions for iodine and cesium that were where the attenuation factor compared with WASH-1400 was much less than a factor of ten?

A. Yes.

Q. Were you not?

A. Yes.

Q. In your testimony, you state that

insofar as IPPSS relies upon the WASH-1400 source terms, it might, by doing so, be conservative in its estimates. Do you recall that? I believe it might be page 3.

A. Yes, yes, I have that, and I recall that.

Q. Are you aware that IPPSS did not, in fact, use the WASH-1400 source term estimates but reduced them in several instances?

A. Well, I understand that the, what we call the WASH-1400 behavior model, where one uses a rather simplistic description of physical behavior in the reactor coolant system, and then a code like CORRAL to describe aerosol precipitation or depletion in the containment.

That portion of it, I believe, they used the WASH-1400 model or source term. Then, of course, my understanding is that they did a great deal in plant specific containment event analysis where you are dealing with a containment different from the PWR containment in WASH-1400.

I would still describe that as a WASH-1400 model referring, of course, to the physical chemical industry model for release from the reactor coolant system and for behavior in containment.

Q. But do you recall whether in the source term fractions that IPPSS used they imposed some reductions over, some reductions as compared with those used in WASH-1400?

A. I would expect so. We have to be careful of terminology here. Normally, when we speak of the source term fraction or source term, we mean what gets out with the accident culmination, namely, the containment failure occurs, so that since you are dealing with a different containment in this different plant, you might, indeed, have a different or quantitatively different source term, but nevertheless, the physical chemical industry description within that containment and within that reactor coolant system might still have been the same as was done in WASH-1400.

So technically, the source term would have a different number, but the same model.

Q. Do you recall in IPPSS that there was a probability distribution where at one end, it was either the WASH-1400 estimates or a factor of two above those were used and then as you went down the spectrum, there was a factor of half times the WASH-1400 estimates and then a tenth of the WASH-1400 estimates?

A. I don't recall that specific thing. I have read portions of the IPPSS study, and I'm not expert to testify in exactly how they treated things there.

Q. Well, then, just to be clear on your testimony, the import of it is that if IPPSS had used the WASH-1400 model and estimates unadultered, it would in the process, be somewhat conservative, but you take no position on IPPSS as it exists, whether it, in fact, it has treated source term conservatively?

MR. BRANDENBURG: I object to the form of the question, Mr. Chairman.

JUDGE GLEASON: Objection is overruled. Answer the question.

A. I take it rather the thrust of my testimony is that insofar as the IPPSS used the physical chemical industry models of WASH-1400, with respect to radionuclide behavior in the reactor coolant system and in containment prior to containment failure insofar as the IPPSS used those models, I would describe it as a conservative use or an overestimate of source term.

Q. But if IPPSS, in addition to using those models, reduced the numbers by a factor of five or a factor of three or a factor of ten, that would have an offsetting effect, would it not.

MR. BRANDENBURG: Object to the form of that question, Mr. Chairman.

JUDGE GLEASON: Objection denied. Answer the question.

A. I have no testimony or expert knowledge on further alterations or modifications of those models.

Q. So you express no opinion as to whether IPPSS in its totality has treated the source term issue in a conservative manner; is that correct?

MS. MOORE: This was asked and answered, this question. I object.

JUDGE GLEASON: There may be an answer but he's scurying an answer, Miss Moore.

Now, the question was does he in fact know it was in the IPPSS in this area or doesn't he. If he says he doesn't, then we can -- which I thought he said at one point, but he keeps throwing caveats in, so I don't know where he stands.

A. As I understand IPPSS, it takes the WASH-1400 model of radionuclide behavior in the reactor coolant system and in the containment prior to containment failure and then after some analysis of containment, draws some conclusions about delayed overpressure failure of containment as being a predominate containment failure mode.

I have not reviewed and cannot -- I don't know in detail what they did to justify the delayed containment.

The only part of that that I address is their model for radionuclide behavior in the reactor coolant system and in containment, and that was all I was speaking to in my testimony.

MR. BLUM: Well, could we possibly have the question read back? I would like an answer to the specific question.

JUDGE GLEASON: All right. Let's have the question read back.

(Question was read back.)

A. That is correct.

JUDGE GLEASON: All right. Fine.

Q. Thank you. You recall speaking earlier about the effect of source term reductions on the value of mitigation measures? Do you recall that?

A. Yes, I recall that.

Q. You are familiar with a general practice of assessing the value of mitigation measures by their relative risk reduction; are you not?

A. Yes, I am.

Q. And a reduction in source term would not necessarily lessen the relative reduction factor of mitigation measures, would it?

A. I think you said "the relative factor." I said earlier that I believe that if you reduce the source term, meaning you reduce the amount of radioactive material which is released from the plant under any given accident sequence, you have reduced the risk of the plant, the estimated risk of the plant, and, therefore, there is less risk to be further reduced by the addition of some mitigative feature.

In that respect, a reduction of source term has the force of reducing the risk reduction effectiveness of a mitigative feature.

Q. Yes, I understand that as regards absolute risk, but perhaps if I give an example, if you assume that Indian Point has a certain level of risk and then with a system of filtered vents it's level of risk is reduced by a factor of ten and then you assume Indian Point with a lower source term, the lowering of the source term would not necessarily mean that the relative reduction value of the filtered vents would no longer be ten, would it?

A. I think it would. I think a lower source term would, indeed, on its face reduce the risk reduction effectiveness of a feature such as a filtered vent.

Q. Wouldn't the relative risk reduction really depend on how the reduced source term affected the importance of various accident sequences as compared with one another?

A. Yes, it would. I am presuming for this discussion that the reduction in source term is a relevant one to this accident scenario or to this mitigative feature. A filtered vent containment system for example, has no merit in reducing the risk of Event V, which is a containment bypass sequence.

I'm assuming in your questions that we are discussing a source term, an accident sequence and a mitigative feature which all have relevance to one another.

Q. Aren't you assuming that the source term reductions apply to those accidents that are mitigated by the feature more than they do to those accidents that are not mitigated by the feature?

A. I'm just assuming that they are relevant, that they do, indeed, apply to that accident sequence and that feature in question.

Q. But if the source terms apply to all accident sequences equally, those which were mitigated and those which were not, would it not have no effect on the relative risk reduction of the mitigation feature?

A. I'm afraid I don't understand the question. Could you repeat that, please. You have at least two negatives in that question. I don't understand it.

Q. Okay. It's true, is it not, that the way relative risk reduction of a mitigation feature is affected is by a differentiation effect on the accident sequences that are mitigated and the accident sequences that are not mitigated? Do you follow that?

A. Yes, I do.

Q. So that if you had a reduction in source term that hypothetically apply to all sequences equally, those that were mitigated and those that were not --

A. Yes.

Q. -- that reduction in source term would not affect the relative risk reduction provided by the mitigation feature?

A. No, I think it would. If I take your definition that, let's assume we have a reactor with only two accident sequences and unit source term applying to each accident sequence and we reduce hat source term to .5 for each accident sequence and I have a mitigative feature that can operate only on one of the two accident sequences, that mitigative feature now has only .5 of what it had to work on before, and it is inherently reduced in effectiveness because there is less risk reduction for it to perform.

Q. That's true for the absolute consequences, the number of people who would be killed or the amount of property damage, but it's not true for the percentage of risk that's alleviated by the mitigation feature, is it?

A. If you are saying that, now at .5 for each source term a mitigative feature can virtually eliminate one of the .5s and, therefore, have the risk, yes, I agree with you, if that's how you define the relative risk.

Q. Isn't that generally how relative risk is defined in considering mitigation features?

A. Not really. Then I think we have a more fundamental disagreement of thought. The risk reduction effectiveness of a mitigation feature is generally spoken of as the risk reduction compared to some cost or penalty for putting in such a system; and the risk reduction is measured as risk before minus risk after, where it is, in a sense, the increment of total risk or the increment of absolute risk and not merely the percentage.

The percentage can be quite misleading. 50 percent of a very small number is not significant. 50 percent of a very large number, of course, is significant.

Q. Okay. I think we understand each other now.

As to whether the approach you first mentioned being the approach that's generally accepted, there will be some further discussion with another witness, but we can go onto another topic.

You stated in your cross-examination by the the Licensees that with regard to using conservative source term estimates, it was very important for any decision-maker that used them to understand the uncertainty and the biases that were embedded in the numbers they used.

Do you recall that?

A. Yes, I do.

Q. And would you stand by that same principal as regards all use of quantitative numbers by decision-makers?

A. Yes, I do.

Q. So that would apply, for example, to uncertainties and biases in a calculation accident probabilities as well as to source term?

MS. MOORE: Objection, Mr. Chairman. We are getting beyond the scope of the witness's direct testimony. He's here to testify about the Staff's position concerning a particular given subject.

MR. BLUM: Well, that's the last question in the line. In essence, it was answered by the previous one. I don't think it's terribly beyond his testimony, but it's not worth a fight.

JUDGE GLEASON: I was going to say all the cross-examiners have gone way beyond the witness's direct testimony.

Answer the question.

A. Yes, in risk analysis overall, probabilities and consequences, I think, it is important for the decision-maker to know the uncertainties.

Q. Could you elaborate for us, in your view, the relevance of the wind scale accident for Indian Point? What similarities do you perceive between the plant with wind scale and Indian Point Plants?

A. Based on my limited experience or knowledge of the wind scale accident, I see very little similarity, except, perhaps, in some individual physical chemical processes within the core. That was a gas-cooled reactor which essentially had a core fire, and though a physical chemist might find something relevant to Indian Point, I see little similarity.

Q. What differences are there in the plants that make it difficult to exstrapolate results from one to the other?

A. It was a different type reactor. It was a gas-cooled reactor with a graphite core. It had an accident mode which cannot happen physically in a pressurized water reactor. The core cannot burn in a pressurized water reactor in place, that is.

Q. Okay. Could you now tell us some, essentially the same question for SL-1, what similarities are there between that reactor and Indian Point?

A. SL-1 was a water reactor. It had a peculiar accident sequence, a power excursion or power burst from shutdown, and it has some similarities to accidents that might occur in a pressurized water reactor such as Indian Point.

Q. What differences are there that make it difficult to exstrapolate results from SL-1 to Indian Point?

A. Well, the accident sequence in particular, the power burst the from full shutdown as against a coremelt accident following operation.

Q. Why does that different significantly for making calculations with regard to source term expectations?

A. Well, the behavior is going -- in

the SL-1 accident sequence, relatively explosive release of material. The fission process was started and went so rapidly that it went out of control and caused what amounted to a steam explosion, which broke up the reactor to a very great extent, and that is physically a very different sequence of events than having some sort of, oh, say, a blackout sequence where there's a loss of power and the core is not cooled and slowly heats up and melts.

Q. Why would these physical differences in the sequence be likely to affect calculations with regard to source term?

A. Different distributions of radionuclides. You have different heat transfer, different mass transfer patterns.

Q. When you say "different," can you give us some sense of how large a difference you are talking about?

A. No, I cannot.

Q. But I presume you don't mean these are simply small differences?

A. Oh, they are major differences in a scenario, and one can look at an accident sequence like SL-1 and learn something about the behavior of, say, iodine, under those accident circumstances that may be useful insight into the behavior of iodine in other sequences such as the ones I spoke of or we speak of in large commercial power reactors.

Q. Well, you wouldn't endorse quantifications based on comparison with that plant?

MR. BRANDENBURG: I'll object, Mr. Chairman. I don't know what quantifications the question assumes.

MR. BLUM: Well --

JUDGE GLEASON: Answer the question.

A. If one is quantifying a model for the behavior of some radionuclide based on some data from that accident, it could be a valid basis for quantification.

If one took simply the release from such an accident and tried to characterize that as a fair quantification of the release from a totally different accident in a light water power reactor, I don't think that would be valid.

Q. Well, when you say, "it could be," is it your testimony that it's plausible to you that it might be but you do not know whether it is?

A. Yes, that's right.
Q. Now, if you could also address the Fermi Breeder Reactor, what similarities are there between that and Indian Point.
A. That was a metal core, liquid metal cooled breeder reactor and it's a different type of reactor. The mechanisms at work were quite different than what we have in a light water reactor accident.
Q. So is it your testimony that it would be very difficult to exstrapolate results from that type of reactor to Indian Point?
A. Yes.
Q. Could you tell us a little bit about the most recent and accurate time table for completion of NRC sponsored source term research?
A. As I said in my testimony, I think on the closing pages, we are at the present time doing analyses of five plants and preparing for each of these plants a detailed physical model of radionuclide behavior in the dominant accident sequences, and by " dominant," I mean the ones selected to cover the full spectrum of accident types, not by their probability but by their physical behavior.

We are doing five plants, Surry, Peach Bottom, Grand Golf, Sequoia and Zion, and we expect to have the five analyses complete this summer; and, in addition, we are preparing a separate -- or our contractor is, Oak Ridge National Laboratory is preparing a separate report to provide a fairly concise definition of the technical data base for these detailed code predictions.

That report also is expected or scheduled to be available late this summer.

With those combined reports of what I'll call the physical science of source term prediction in hand, we will be doing a peer evaluation through the later months of this year, and we expect by the end of the year to be in a position to judge whether we, indeed, have a sound basis for quantitative reassessment of severe accident source terms.

Q. Thank you.

MR. BLUM: I have no further questions.

JUDGE GLEASON: Any redirect?

MS. MOORE: I have no redirect, your Honor.

JUDGE PARIS: Mr. Bernero, will you describe the pathway for the release of iodine in the TMI 2 accident briefly?

MR. BERNERO: My description of this is based on my having participated in the TMI special inquiry for the US Nuclear Regulatory Commission, the so-called Ragovan inquiry.

As I recall, the accident sequence early on had a bypass from the reactor building sump into the auxillary building where some water of relatively low activity level got into the auxillary building and out to the floor.

It is my belief from the studies we did in that inquiry that the releases from that amount of water are negligible or were negligible.

The iodine release in TMI was predominately, I believe, from the letdown and makeup system of the reactor, which was operating throughout the early days of the accident, namely, taking reactor coolant from whatever the pressure was at the time, reactor coolant system pressure, bringing it out into the auxillary building through the letdown oriface, depressurizing it and putting it into holding tanks and later pumping it back into the reactor coolant system.

During the degasification, which accompanied this depressurization, noble gasses, in particular, were stripped from that coolant. They normally would have gone through a body of piping called the vent header into waste gas delay or hold tanks.

The vent header was leaking at the time, and virtually every time there was a letdown and pumpover of this gas, there was a leak. It would leak from the header and go out the stack in the ventilation system from the auxillary building.

The activity so releaseed was detectable principally because of the radioactive noble gasses, and I believe that the iodine that did get out came out along with that, a small amount of iodine at that same time or those same releases.

JUDGE PARIS: Well, is it correct that most of the iodine that was released at TMI 1 was not released as molecular iodine?

MR. BERNERO: I don't know in what form -- I don't know in what form it was released, whether it was as organic iodine or molecular. I believe it would have been a volatile species to travel the way it did, with the noble gasses, either the organic iodides or molecular iodine, more likely the organic.

JUDGE PARIS: You said in your testimony that the TMI 2 accident might not be a suitable model for a containment breach type accident as far as behavior of iodine is concerned. Could you elaborate on that a little bit?

MR. BERNERO: The iodine that would be released in a containment breach accident would be whatever iodine is readily available in the atmosphere of the containment at the time of containment breach.

If, on the other hand, one would take the water from the reactor as was the case in TMI and slowly circulate it out into another building so that the noble gasses are vented off of it, but if there is no massive force to release iodine from it, I [illegible]n't think you get a fair estimate of the am[illegible]at could be released.

It comes much, much later in the accident. The release in TMI occurred over a period of several days in the case of TMI, and the containment breach accidents of risk significance are ones where the containment would be breached within hours of the onset of the accident long before there are chances to dissolve the iodine or other species with sprays, with water, with whatever is available or was available at TMI.

The TMI accident, as I tried to say in my testimony, gives us useful insights into what happened inside the core, what was released from the core into the reactor coolant and then what was released from the reactor coolant into the containment atmosphere, but for high-risk sciences, for risk significance, all of that information in the first few hours is what counts, the first few hours and not over that period of many days.

JUDGE PARIS: Okay. That is very helpful. Thank you.

(Continued to next page; no context lost.)

JUDGE SHON: I had a couple of questions, Mr. Bernero, generally focused on the notion of CORRAL Code and the difference from Stratton Rodger.

In particular, on page 7 of your testimony you suggest that the CORRAL Code already gets quite a bit of crib for the solubility of iodine in water, and you mentioned this very large number, the concentration in the molecular phase is more than 10,000 times higher than that in the containment atmosphere. That's of the nature of a partition factor.

THE WITNESS: Yes.

JUDGE SHON: It is not a real measure of quantity in either phase because you have to know the relative volume of the phases, and that sort of thing.

So that could mean that it still shows quite a bit of iodine in the atmosphere, doesn't it?

THE WITNESS: It could. The point I was trying to make is that if you go into WASH 1400, Appendix 7, on physical process, it describes the way it derived its model, which we call the CORRAL Code, as an empirical fit to the data from the containment safety experiment series, and the point of that partition factor being high was such that it was so high that it even led to controversey at the time that it claimed more solubility for iodine.

Mind you, it was treating iodine as elemental iodine. It claimed more solubility than was deemed appropriate by many peopl: at the time. It was too optimistic, too much reducing the source term.

The parameters are such that that partition factor will give you excellent fission product reduction for iodine, excellent attenuation if you have almost any spray system operation.

As I said in the testimony, the WASH 1400 concluded that the reactor building sprays are very effective engineering safety features because of that.

JUDGE SHON: Do you have any idea what the corresponding partition factor might be if the species being considered is cesium iodide instead of iodine?

THE WITNESS: I think it would be much higher. I wouldn't know a number to put on it.

JUDGE SHON: I would have guessed that too.

THE WITNESS: Actually, it is quite different now. It is not a solubility equilibrium. Now it is a salt solubility.

JUDGE SHON: Salts are notoriously soluable?

THE WITNESS: Yes.

JUDGE SHON: And they are of a very low volatility, generally speaking.

The second question touches on something that you mentioned a moment ago and it is on pages 8 and 9 of your testimony. I may be asking you to explain something someone else has done that you don't really find yourself in sympathy with. But you mentioned that the CORRAL Code is really based on the CSE, at the bottom of this page 8 and top of page 9.

And then you say that the offers of the Stratton testimony reject the quantitative prediction of the radionuclide behavior in the containment provided by CORRAL because of their belief in the CSE results.

I don't quite understand why that

should be. If they like the CSE and CORRAL likes the CSE, then why don't they seem more like one another?

THE WITNESS: You are reitterating our dilemma as well. It is a matter of clarification. When we at NRC studied their testimony, it appeared to us that they were not recognizing that the CORRAL Code is basically an empirical fit of the CSE data, and we were somewhat puzzled. They seemed to be rejecting the code but not the data source. It was a lack of clarity, and that was the point we were trying to make here, that if you accept the one, you accept the other, unless, of course, you say it wasn't properly fit, that there was a technical error in the way the data were interpreted.

JUDGE SHON: So you are just saying you don't quite understand?

THE WITNESS: I don't quite understand their apparent criticism of CORRAL and yet at the same time citation of CSE as a good source of data.

JUDGE SHON: The last is a fairly fundamental question and perhaps I should have addressed it to you earlier. It certainly forms an earlier part of your testimony today and it again relates to cesium iodine and cesium iodide, and it has to do with the grounding which you say is generally accepted for the notion that the chemical form of the cesium and iodine, or at least of the iodine, is a cesium iodide.

This is based, in part, I believe you said, on the chemical thermodynamics of cesium and of iodine, is that correct?

THE WITNESS: Yes. Free energy calculations indicate that conditions are favorable for that compound to form.

JUDGE SHON: Now, these are elements that are present, radio elements that are present in what I will call carrier free form. It is notorious in physical chemistry that when the concentrations of things are as low as they are, if you carry the free form, ordinarily chemical thermodynamics may be defeated by other factors. You are aware of that.

How do we know that here this sort of thing isn't being interfered with?

THE WITNESS: I am afraid I have to defer to people far more expert than I am in this field. I think that that was one of the reasons the investigators in WASH 14000 looked at cesium iodide and gingerly set it aside and chose to treat it as free agents, the more mobile species.

I really here on the collective judgment of all of the experts who are contributing to our work now and who have contributed to NUREG 06772, published in 1981, where they have addressed these issues and concluded with general consensus that one can trust the form to be predominantly cesium iodide for the iodine, and cesium hydroxide, I believe, for the bulk of the other cesium available.

There is some residue due of organic iodide still to be accounted for.

JUDGE SHON: Thank you. I have no further questions.

JUDGE GLEASON: Thank you, Mr. Bernero. You are excused.

Will the licensees proceed with their panel, please.

Whereupon

DONALD PADDLEFORD, was sworn by the administrative judge, and testified as follows:

MR. BRANDENBURG: Mr. Chairman, at this time the Power Authority and Con Edison would like to call to the stand Dennis C. Bley, Thomas Potter and Dennis E. Richardson.

Whereupon

DENNIS C. BLEY

DENNIS RICHARDSON

THOMAS POTTER, having been previously sworn, testified as follows:

DIRECT EXAMINATION

BY MR. BRANDENBURG:

Q. Will you state your full name and address for the record, please, Dr. Bley?

A. (Witness Bley) Dennis C. Bley, Pickard, Lowe & Garrick in Oakland, California.

Q. Mr. Paddleford?

A. (Witness Paddleford) Donald F. Paddleford, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

Q. Mr. Potter?

A. (Witness Potter) Thomas E. Potter, Pickard, Lowe & Garrick, Washington, D. C..

Q. And Mr. Richardson?

A. (Witness Richardson) Dennis C. Richardson, Westinghouse Electric Corporation, Pittsburgh, PA.

Q. Do each of you have before you a copy of a document entitled "Licensee's testimony of Dennis C. Bley, Donald F. Paddleford, Thomas E. Potter and Dennis F. Richardson on commission question five?

(All answer in the affirmative)

Q. And do each of you also have before you a copy of a letter signed by Mr. Colarulli and myself dated April 1, 1983, to the licensing board attached to which is an errata sheet with four entries and a replacement at table 2, which is at page 11?

(All answer in the affirmative)

Q. Were each of these documents prepared by you or under your direct supervision?

(All answer in the affirmative)

Q. Other than the changes reflected in the errata sheet, do you have any further additions to make to these documents at this time?

A. (Witness Bley) Yes, we have three minor corrections.

Q. Identify them, please.

A. (Witness Bley) On page 22 of the testimony, under item 8, at line 6, delete "either of two" and replace it with "any of three." It reads "any of three component cooling water pumps."

On page 24, the second paragraph under section 6, "Conclusions," the second line of that paragraph, insert the words "health risk" after the word "preliminary," so it reads "preliminary health risk safety goals."

And in Donald F. Paddleford's statement of professional qualifications the second item, postgraduate course in engineering, it should read "CMU, not "SMU."

That's all.

Q. With these changes, errata and additions is this testimony true and accurate to the best of your knowledge, information and belief?

(All answer in the affirmative.).

Q. Do you adopt it as your testimony in this proceeding?

(All respond in affirmative."

MR. BRANDENBURG: Con Edison and the Power Authority move the admission of this testimony in this proceeding and ask that it be bound into the record as if read.

MR. BLUM: We do have objection to section 3 of this testimony and we will be moving to strike.

JUDGE GLEASON: What page is that?

MR. BLUM: That's on page 4 and it is titled "Comparison with the Nuclear Regulatory Commission preliminary safety goals." Our motion to strike is based on two separate grounds.

JUDGE GLEASON: Which pages does it cover now? All of section 3?

MR. BLUM: Yes. It would cover pages 4, beginning six lines down, through page 8.

JUDGE GLEASON: Let's hear your objections.

MR. BLUM: We would also be moving to strike six lines out of the conclusion which reiterate this portion of the testimony.

The first ground for the motion to strike is that the document which the licensees cite, that is, what is put out by the Nuclear Regulatory Commission on Monday, March 14, 1983, entitled, "NRC sees public comments on plan for evaluating safety goals," specifically includes an instruction that these safety goals are not to be used in hearings or in licensing process.

Specifically, on page 7, the bottom paragraph reads, "The qualitative safety goals and quantitative design objectives contained in the commission's policy statement will not be used in the licensing process or be interpreted as requiring probable risk assessments by applicants or licensees during the evaluation period. The goals and objectives are also not to be litigated in the commission's hearings. The staff should continue to use conformance to reglatory requirements as the exclusive licensing basis for plants."

The basis for this statement is that these goals are preliminary at the present time, they are put out for comment, for general guidance, but they are not to be given any weight in the hearing processes, and, therefore, any reliance on them by a licensing board would be premature at this time and has been specifically forbidden by the commission.

I can go on and mention our second basis for the motion to strike.

JUDGE GLEASON: Why don't we get it all in.

MR. BLUM: The second basis is there is really no probative value to this portion of the testimony and it is apt to be somewhat confusing. The probative value is measured with reference to the commission's question which calls for a comparison of the Indian Point risk with the range of risks of other existing nuclear plants.

Now, the safety goals are not derived from any summary of risks of existing plants averaged together. They are picked on the basis of comparison with other sorts of accidents or mortality outside the nuclear area, or whatever. But it does not, it absolutely does not represent the range of risks of other nuclear plants.

Now, it is conceivable that the two m y coincide, for whatever reasons the safety goals may or may not be the same as the risks of most other plants. But the only way that would be known is by comparison with the other PRAs.

That is, what would really be done is to compare the safety goals with the other PRAs so the safety goals could then be used as a range of risks and then comparing the safety goals. But if that's what's being done, this really adds nothing over comparison with the other PRAs, which is what is done in section 4 of this testimony, and is really the relevant part of the testimony.

So section 3 really adds nothing in the way of answering the commission's question and is simply apt to be a source of confusion.

JUDGE GLEASON: Do you have a third objection or is that it?

MR. BLUM: Those two are it.

JUDGE GLEASON: Mr. Brandenburg.

MR. BRANDENBURG: I would like to speak first, Mr. Chairman, and perhaps Mr. Colarulli has some other grounds.

I think that the motion should be denied if for no other reason that Mr. Blum has failed to heed the board's advice to advise the parties in advance of intentions to strike.

Prior to hearing Mr. Blum's remarks just a few seconds ago, he at no prior time communicated to me his intention to strike this testimony, although he has had it since March 22.

Now, just listening to Mr. Blum's words, I think they the argument contains the seeds of its own lack of merit, Mr. Chairman. Even as Mr. Blum referred to the commission's statements which accompanied the issuance of the draft safety goals, Mr. Blum himself quoted words to the effect that this document should not be used in the licensing process, and I hasten to add that this is not a licensing process This is not a licensing hearing.

These plants have a license and that license is not something that is being litigated here.

The commission quite clearly intended that the safety goals should not be used as a de facto speed limit or green light or anything for licensing in the absence of all of the other complex licensing considerations that are attendant to such a proceeding.

This is not such a proceeding and I submit that absolutely no hardship is being or lack of respect is being given to those instructions of the commission by virtue of the fact that this is a mere investigatory proceeding.

As far as the lack of probative value of these commission goals, I think even Mr. Blum would be the first to admit that a lot of thought, very careful thought within the commission and the commission staff went into these.

While they are indeed preliminary, nonetheless I think that they do offer valuable insights at this point in time for this licensing board to make analogies, and so on, and to reach its conclusions with respect to commission question 5.

To the extent that the probative value of these goals is less than because they do not specifically reference other plants, I think that is a factor that this board is fully taking into consideration as it weighs the testimony.

MR. BLUM: Your Honor, if I may respond to Mr. Brandenburg's first point, I did notify two attorneys for the Power Authority and had assumed that that would get passed on to Mr. Brandenburg.

JUDGE GLEASON: When did you do that?

MR. BLUM: At the first opportunity this morning. Last night was the time when we began to make the decision to make the motion to strike on this testimony.

MR. BRANDENBURG: That testimony sounds self contradictory, Mr. Chairman.

JUDGE GLEASON: Do you have anything to add, Mr. Colarulli?

MR. COLARULLI: Just briefly, your Honor. This clearly is a unique proceeding. The commission set a mandate for this board to determine what is the risk posed by the plants and how does that risk compare to other risks posed by other plants.

Clearly when the commission did that in January of 1981, and again in September 1981 in its orders, it did not have before it the March 14, 1983 preliminary safety goals. One would posit, going back to the commission and saying: Commission, do you want us to look at the safety goals? But clearly, without doing that, this is a goal, a measuring stick that could be used to great effect in this proceeding.

I would note that staff in its question 5 testimony has also, with a number of qualifications as we have, used it in some way to measure the risk proposed by Indian Point. I know that UCS's witness, Mr. Shalley, makes several passing negative comments about the safety goal as well.

So, clearly all the parties to some extent have addressed the safety goal. We believe the commission should not be denied this valuable piece of information.

JUDGE GLEASON: Does the staff have some comments?

MS. MOORE: Yes, your Honor. While we

do agree that the policy statement does say that the safety goals themselves are not to be litigated in licensing proceedings and the goals are not to be used in licensing proceedings, we, too, believe that in this particular proceeding for the limited purpose of presenting perspectives on the risks posed by Indian Point, that a mention of the safety goal is appropriate, and in fact we have done just that in our question 5 testimony, for the purpose of presenting some perspectives rather than for a comparison purpose or for using the safety goal as a reglatory tool, as I believe our testimony would state.

MR. BLUM: If I may respond, it is unclear -- there is talk about general perspectives being provided but it doesn't seem to be an answer to the commission's question, which asks for a comparison of this plant and other plants in general, and the safety goals are decisively not that.

The other thing is I would mention that in those respects it is being treated as a licensing hearing. There are many aspects of rules and procedures of licensing hearings that have been incorporated for this one and it is a hearing where the license of a plant is at stake.

It is somewhat different than usual where the plant begins without a license and it is determined whether it will acquire one. But it is a hearing oriented ultimately toward the question of a license.

JUDGE GLEASON: Your point being it would be more important to have fixed standards in this kind of proceeding than even in regular licensing proceedings? On safety goals I mean.

MR. BLUM: I am sorry, I didn't understand.

JUDGE GLEASON: Let it go, Mr. Blum.

(There was a pause in the proceeding.)

JUDGE GLEASON: I guess you can gather somewhat by the pause that the board has a little bit of uncertainty with respect to this motion and to the references of this testimony and the Nuclear Regulatory Commission's preliminary safety code.

We have concluded that our best course is to deny the objection, to let the testimony in, with the clear understanding that we will be taking a very careful look at it, as well, in fact, all other aspects of the study itself, as we make our recommendations as to what kind of reliance, if any, we are going to put on it.

That's about where we are.

All right.

MR. BRANDENBURG: Mr. Chairman, the panel is ready for cross-examination.

JUDGE GLEASON: With that objection denied, the testimony will be admitted into evidence and bound into the record as if read, with the errata sheet and other changes indicated.

(The bound testimony follows.)

JUDGE GLEASON: Do you wish to proceed, Mr. Blum?

MR. BLUM: Certainly.

CROSS-EXAMINATION

BY MR. BLUM:

Q. Gentlemen, could you identify whether there are any parts of the testimony that you, as individuals, wrote to help address the questions?

A. (Witness Bley) Yes, we can, to some extent.

Section 5, the special design features should, for the most part, be directed to Mr. Paddleford.

The questions that deal with

uw #1020

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

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

In the Matter of	)	
	)	
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point, Unit No. 2)	)	Docket Nos. 50-247 SP 50-286 SP
	)	
POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3)	)	March 22, 1983

LICENSEES' TESTIMONY OF DENNIS C. BLEY, DONALD F. PADDLEFORD, THOMAS E. POTTER, AND DENNIS C. RICHARDSON ON COMMISSION QUESTION FIVE

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I. PRESENTATION AND QUALIFICATIONS OF PANEL MEMBERS

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

My name is Donald F. Paddleford. I am an Advisory Engineer in the Risk Assessment Section of the Nuclear Safety Department of the Nuclear Technology Division of Westinghouse Electric Corporation. I was a principal investigator on the Indian Point Probabilistic Safety Study. A statement of my professional qualifications is attached.

My name is Thomas E. Potter. I am a consultant at Pickard, Lowe and Garrick, Inc., in public health consequence analysis of radioactive releases. I was a principal investigator on the Indian Point Probabilistic Safety Study. A statement of my professional qualifications is attached.

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

II. INTRODUCTION

A central issue in this hearing is whether the Indian Point nuclear power plants produce risks that significantly exceed the range of risks posed by other nuclear power plants in light of the demographic characteristics of the area surrounding the Indian Point site. This issue is articulated in Question 5 of the Commission's Memorandum and Order of January 8, 1981, which asked:

> 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.)

Risk can be measured by several health and economic indices and from both an individual and a societal standpoint. Population distribution and plant characteristics affect these indices differently. In selecting which indices are most important, guidance is taken from the Nuclear Regulatory Commission's (Commission's) preliminary safety goals, which emphasize early and latent fatality risks (Reference 1). Similarly, the emphasis here is on the effects of population distribution on early and latent fatality risk.

Three different approaches to addressing Commission Question 5 are taken in this testimony. First, a comparison is made of the risks from the Indian Point plants to the Commission's preliminary safety goals. Second, the risks from the Indian Point plants, as analyzed in the Indian Point Probabilistic Safety Study (IPPSS) (Reference 2), are compared to the results of site and plant specific probabilistic risk assessments (PRAs) of a number of other nuclear power plants. Third, there is a discussion of the benefits resulting from the special design features at Indian Point which are not present at all nuclear power plants.

Individually and collectively, each of these comparisons supports the conclusion that the Indian Point nuclear power plants are in the range of risks posed by other nuclear power plants. Specifically, (1) the risk of latent fatalities at Indian Point is low and information available suggests that latent fatality risks may not vary greatly among nuclear power plants; (2) the absolute risk of early fatalities is even lower than the latent fatality risk, thereby reducing the significance of plant-to-plant variability; (3) for both risk indices, the Indian Point plants meet the Commission's preliminary safety goals; and (4) anticipated reductions in source term estimates would reduce both early and latent fatality risk and, in fact, could effectively eliminate the early fatality risk. <u>See</u> Licensees' Testimony of William R. Stratton, Walton A. Rodger, and Thomas E. Potter on Question One (Jan. 24, is made of the risks from the Indian Point plants to the Commission's preliminary safety goals. Second, the risks from the Indian Point plants, as analyzed in the Indian Point Probabilistic Safety Study (IPPSS) (Reference 2), are compared to the results of site and plant specific probabilistic risk assessments (PRAs) of a number of other nuclear power plants. Third, there is a discussion of the benefits resulting from the special design features at Indian Point which are not present at all nuclear power plants.

Individually and collectively, each of these comparisons supports the conclusion that the Indian Point nuclear power plants are in the range of risks posed by other nuclear power plants. Specifically, (1) the risk of latent fatalities at Indian Point is low and information available suggests that latent fatality risks may not vary greatly among nuclear power plants; (2) the absolute risk of early fatalities is even lower than the latent fatality risk, thereby reducing the significance of plant-to-plant variability; (3) for both risk indices, the Indian Point plants meet the Commission's preliminary safety goals; and (4) anticipated reductions in source term estimates would reduce both early and latent fatality risk and, in fact, could effectively eliminate the early fatality risk. See Licensees' Testimony of William R. Stratton, Walton A. Rodger, and Thomas E. Potter on Question One (Jan. 24, 1983). In addition, plant features present at Indian Point but not included at other plants are among the important factors supporting the conclusion that the Indian Point nuclear power plants are within the range of risks posed by other nuclear power plants.

III. COMPARISON WITH THE NUCLEAR REGULATORY COMMISSION'S PRELIMINARY SAFETY GOALS

On March 14, 1983, the Commission published a Policy Statement on Safety Goals for the Operation of Nuclear Power Plants. 48 Fed. Reg. 10,772 (1983). The preliminary safety goals and design objectives apply to both individual and societal risk. Subordinate to these goals is a design objective for risk to the plant, core melt frequency.

The preliminary safety goals represent a national benchmark against which all nuclear power plants can be compared. Therefore, the comparison of the risks from the Indian Point plants to the Commission's preliminary safety goals is one method of determining if these plants are within the range of risks posed by other nuclear power plants. Both Indian Point Units 2 and 3 are among those plants which have health risks smaller than those adopted by the Commission's preliminary safety goals.

Uncertainties in the calculated health risks for Indian Point are offset by the large margins between these calculated risks and the preliminary safety goals. Reduced source terms will result in even larger margins.

A. Individual Risk

The Commission's preliminary safety goals state that the early fatality risk to an average individual in the vicinity of a nuclear power plant[1] should not exceed one-tenth of one percent of the sum of early fatality risk to that individual from other accidents. 48 Fed. Reg. 10,774.

To translate this goal into numerical form, we use the United States national average accident risk of 5 fatal accidents per 10,000 people per year (5×10^{-4} per year) (Reference 2).

For the purpose of assessing the individual risk, the Commission defines "vicinity" of the plant as a 1-mile radius. 48 Fed. Reg. 10,774. Using this definition of vicinity and IPPSS emergency response assumptions, the average individual early fatality risk has been calculated

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1. According to the Commission,

> the average individual in the vicinity of the plant is defined as the average individual biologically (in terms of age and other risk factors) and locationally who resides within a mile from the plant site boundary. This means that the average individual is found by accumulating the estimated individual risks and dividing by the number of individuals residing in the vicinity of the plant.

48 Fed. Reg. 10,774.

as a fraction of the national average accident risk. This is then compared with the Commission's preliminary safety goal in Table 1. The risk of Indian Point is well within this goal, by a factor of approximately 70 for Indian Point Unit 2 and a factor of approximately 75 for Indian Point Unit 3.

B. Societal Risk

For societal risk, the Commission's preliminary goal is that the latent cancer fatality risk to the population in the vicinity of a nuclear power plant should be less than one-tenth of one percent of the cancer fatality risks from other causes. 48 Fed. Reg. 10,774. For latent fatalities, vicinity is defined as 50 miles. Id. The national average cancer risk for a person in the United States is two deaths per 1,000 people per year (2×10^{-3} per year) (Reference 2).

For this radius from the Indian Point plants, the average latent cancer fatality risk has been calculated as a fraction of the national cancer fatality risk and is compared with the Commission's preliminary goal in Table 1. The risk of Indian Point is well within this goal, by a factor of approximately 165 for Indian Point Unit 2 and 710 for Indian Point Unit 3.

C. Core Melt Frequency

Table 1 also shows the comparison of the Indian Point Units 2 and 3 median core melt frequencies with the Commission's preliminary safety goal. Because the Zion PRA

TABLE 1

COMPARISON OF RISKS FROM INDIAN POINT PLANTS WITH NRC SAFETY GOALS

	Indian Point 2	Indian Point 3	NRC Goal
Average Early Fatality Risk Within 1 Mile as a Fraction of Other Accident Fatality Risk Within 1 Mile	$1.4 \times. 10^{-5}$	1.3×10^{-5}	1×10^{-3}
Average Latent Cancer Fatality Risk Within 50 Miles as a Fraction of Other Cancer Fatality Risks Within 50 Miles	6.0×10^{-6}	1.4×10^{-6}	1×10^{-3}
Core Melt Frequency (per year of reactor operation) internal plus external	1.4×10^{-4}*	5.0×10^{-5}*	1×10^{-4}
Core Melt Frequency (per year of reactor operation) internal only	5.0×10^{-5}*	3.0×10^{-5}*	no explicit goal stated

*Median Frequency

(Reference 3) and the IPPSS are the only risk assessments of which we are aware that give comprehensive treatment to external events, Table 1 also includes the median core melt frequency of internal initiating events only. Considering internal initiating events only, both Indian Point plants meet the Commission's preliminary safety goal.

Although information on core melt frequency is provided here for completeness in comparing the Indian Point plants against the preliminary goals, the values of this parameter are not of particular use in addressing Commission Question 5. This is because core melt frequency is a poor indicator of public risk, as discussed in Licensees' Testimony on Commission Question One, Board Question 1.1, and Contention 1.1 (Jan. 24, 1983). This can be shown in two ways. First, approximately 65 percent of the postulated core melt scenarios at Indian Point Unit 2 and almost 95 percent of those at Indian Point Unit 3 do not lead to significant releases of radioactive material to the environment. Second, approximately 95 percent of the calculated early fatality risk at each plant is due to the interfacing systems LOCA, which contributes less than one-half of one percent to the core melt frequency. On the other hand, core melt frequency is a useful indicator of economic risk to the customers and owners of Indian Point Units 2 and 3, as it is a measure of the likelihood of losing the benefits of these plants.

IV. PRA COMPARISONS

Another way to compare the risks posed by the Indian Point plants with those posed by other nuclear power plants is to compare site and plant specific PRAs for various plants, all identical in scope and using state-of-the-art methodology. At the present time, however, such a comparison cannot be made due to the limited number of available, comparable studies. The following comparisons, however, are possible:

1. A comparison of the IPPSS risk results with those of other plants for which reasonably complete PRAs have been published. Only the Indian Point and Zion PRAs include external events; therefore, only comparisons on an internal initiator basis have been made.[1]

2. A comparison of the IPPSS results with the range of risks for nuclear plants calculated by the Commission Task Force Report on the Interim Operation of Indian Point (Reference 4).

A. Comparison Of Risks Among Nuclear Power Plants

In connection with the comparison of risks among nuclear power plants, it is important to note that PRA methodology has been evolving rapidly over the last 10 years. The various published studies, therefore, differ considerably in certain respects. Thus, when comparing the

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1. While the Big Rock Point PRA did consider fires, it did not evaluate other external initiating events.

results of IPPSS with those of other PRA studies, it must be recognized that such comparisons are not only of different plants, but are also of different data bases and, in some cases, of different methodologies. These studies vary in scope and sophistication. Some did not include external events and/or public health effects, while others focused only on a few systems or on one type of accident initiator. With these reservations in mind, quantitative comparisons can be made.

1. Individual Risk

Table 2, which was compiled by the Commission Staff (Reference 5), presents data from a number of plant specific PRAs on the frequency of core melt, the frequency of a major release, and the early and latent fatality risks to an individual living within one mile of plant boundaries. The values in the "Early fatality" column can be directly compared with the Commission's preliminary safety goal for this health index (5×10^{-7}). This table generally reflects the range of risks from internal initiating events at United States nuclear power plants because it includes a representative sampling of PWRs and BWRs, high and low population density sites, power levels from 71 to 1250 MWe, and principal reactor vendors and architect engineers. Although this table has been reproduced verbatim from Reference 5, additional information is also presented for Indian Point Units 2 and 3, and appears in a box directly below the

TABLE 2

RESULTS OF EXISTING PROBABILISTIC RISK ASSESSMENTS

*** WARNING - THERE ARE LARGE UNCERTAINTIES ASSOCIATED WITH THE VALUES PRESENTED IN THIS TABLE. ALSO, PRAs WERE NOT PERFORMED USING CONSISTENT METHODOLOGY AND ASSUMPTIONS

	PRA	NSSS/AE	DATE	POWER (MWe)	F CORE MELT 1/ 2/	F MAJOR RELEASE 1/ 3/	INDIVIDUAL RISK WITHIN 1 MILE: EARLY FATAL 1/ 4/	INDIVIDUAL RISK WITHIN 1 MILE: CANCER FATAL 1/ 5/	COMMENTS
ANO-1	IREP	B&W/BECHTEL	81	836	5×10^{-5}	2×10^{-5}	6×10^{-7}	2×10^{-7}	all-PWR 2 8/
Biblis B	GERMAN RSS	FRG (W)	78	1300	4×10^{-5}	1×10^{-6}	3×10^{-8}	2×10^{-8}	Containment stronger and larger than U.S.
Big Rock 6/	WOOD-LEAVER/ SAI	GE/BECHTEL	81	71	1×10^{-3}	0	0	—	Low power level, remote siting 9/
Browns Ferry	IREP	GE/TVA (BWR 4, M I)	81	1067	2×10^{-4}	4×10^{-5}	2×10^{-7}	1×10^{-6}	ATWS and interdependency in redundant RHR trains dominate core melt 10/
Calvert Cliffs	RSSMAP	CE/BECHTEL	82	850	2×10^{-3}	1×10^{-3}	9×10^{-6}	2×10^{-5}	More comprehensive IREP study in progress. AFWS redesign will lower risk and core melt frequency 11/
Crystal River	IREP	B&W/GILBERT	80	825	4×10^{-4}	2×10^{-4}	3×10^{-6}	2×10^{-6}	F core melt reduced by factor of 3 by procedure changes 12/
Grand Gulf	RSSMAP	GE/BECHTEL (BWR 6, M III)	81	1250	4×10^{-5}	4×10^{-5}	1×10^{-7}	1×10^{-7}	Containment always fails directly to atmosphere, does not assume staff's analysis of ATWS risk
I.P. #2 6/	PLG	W/UE&C	82	873	4×10^{-4}	3×10^{-4}	3×10^{-8}	1×10^{-8}	Includes external events 7/ 13/
	PLG	W/UE&C	83	873	1×10^{-4}	4×10^{-8}	7×10^{-9} a	6×10^{-8} a	Includes external events
	PLG	W/UE&C	83	873	5×10^{-5}	4×10^{-8}	6×10^{-9} a	3×10^{-9} a	Internal events only
	a. This is a mean value. The median value would be somewhat lower.								
I.P. #3 6/	PLG	W/UE&C	82	965	9×10^{-5}	3×10^{-5}	1×10^{-9}	3×10^{-10}	Includes external events 7/
	PLG	W/UE&C	83	965	5×10^{-5}	4×10^{-8}	6×10^{-9} a	2×10^{-8} a	Includes external events
	PLG	W/UE&C	83	965	3×10^{-5}	4×10^{-8}	6×10^{-9} a	3×10^{-9} a	Internal events only
	a. This is a mean value. The median value would be somewhat lower.								
Limerick 6/	SAI	GE/BECHTEL (BWR 4, M II)	81	1055	2×10^{-5}	3×10^{-6}	1×10^{-8}	1 x 10[illegible]	Mean value, assumes ATWS fix
Millstone	IREP	GE/EBASCO	82	652	3×10^{-4}	1×10^{-4}	1×10^{-7}	6×10^{-7}	Major release is in Release Category 3
Oconee	RSSMAP	B&W/BECHTEL	80	860	8×10^{-5}	4×10^{-5}	2×10^{-7}	1×10^{-7}	1/4-PWR 2; 3/4-PWR 3
Peach Bottom	WASH-1400	GE/BECHTEL (BWR 4, M I)	75	1065	3×10^{-5}	7×10^{-6}	4×10^{-8}	3×10^{-8}	Staff's analysis of ATWS would likely result in risk exceeding safety goal
Sequoyah	RSSMAP	W-IC/TVA	78	1148	6×10^{-5}	4×10^{-5}	1×10^{-6}	5×10^{-7}	H_2 control reduces risk by 2 to 3
Surry	WASH-1400	W/S&W	75	775	6×10^{-5}	1×10^{-5}	2×10^{-7}	1×10^{-7}	2/3-PWR 2; 1/3-PWR 3
Zion 6/	PLG	W/S&L	81	1100	4×10^{-5}	4×10^{-6}	2×10^{-8}	1×10^{-8}	Includes external events

1/ All numbers are median values or point estimates from internal initiators unless otherwise specified.

2/ Frequency of core melt 1×10^{-4} is the Safety Goal Value for Accident Probability Comparison.

3/ Frequency of release with Potential for Early Fatalities Assuming Nominal Evacuation and Warning Times (RSS).

4/ 5×10^{-7} is the Safety Goal for Early Fatality Risk Comparison. Same assumptions as 3 above unless specified.

5/ 2×10^{-6} is the Safety Goal for Cancer Fatality Risk Comparison. Same assumptions as 3 above unless specified.

6/ Utility-performed PRAs. All values are rough estimates based upon initial interpretation of results.

7/ Optimistic emergency response assumptions (1-hour delay with at least 8-hour warning) for dominant sequence when determining individual risk.

8/ Predicted risk is dominated by small LOCAs and transients. Souce term reduction expected to reduce predicted risk to within guidelines. Likelihood of major release could be reduced by adding parallel valves at the discharge of the borated water storage tank or by improving DC power redundancy.

9/ Low power level (71 MWe) results in low individual risk. Extensive design modifications necessary to reduce core melt frequency.

10/ Reduction of core melt frequency would require redesign of the residual heat removal system to eliminate commonalities between trains which reduce the significance of multiple redundancy.

11/ AFWS redesign is expected to significantly reduce core melt frequency and individual risk. IREP study including improved AFWS design will be available in Spring 1983. Modification to DC power system and engineered safety system actuation system may be required to lower core melt frequency within guidelines. Predicted risk is dominated by transient event and should be significantly reduced by new source term data.

12/ Core melt frequency could be reduced to less than guidelines levels by improving written procedures and improving [illegible] of the [illegible] stage [illegible]. Predicted risk is dominated by small [illegible] events. New source term information is expected to result in a moderate reduction in predicted risk.

13/ Core melt frequency is dominated by seismic considerations.

14/ Core melt frequency could be reduced to below guideline levels by redesigning the emergency AC power system to reduce dependency on the gas turbine and improving procedures for responding to transients. Predicted risk is dominated by transient events. New source term information should result in a significant reduction.

Indian Point results presented by the Staff. This additional information is drawn from the risk calculations in Licensees' Testimony on Commission Question One, Board Question 1.1, and Contention 1.1. It includes risk results from internal initiating events only to avoid an erroneous comparison of Indian Point internal plus external results with internal only results from other plants. It also includes the internal plus external results for Indian Point. Based on the results in this table, the risk to an individual living within 1 mile of Indian Point compares favorably with the estimated risk to individuals living within 1 mile of other nuclear power plants. Additionally, the Indian Point core melt frequency is within the range of other estimates presented in the table, and the frequency of a major release compares favorably with the estimates for the other plants in the table.

Another valuable comparison is the frequency of the interfacing systems LOCA, which is believed to be an important contributor to early fatality risk at PWRs and is the major initiating event contributing to the early fatality risk at Indian Point. Estimates of the frequency of this event at Indian Point and several other nuclear power plants are presented in Table 3. The differences in these estimated frequencies are due to a combination of design differences among plants, as well as to testing and

TABLE 3

COMPARISON OF INTERFACING SYSTEMS LOCA MEDIAN FREQUENCIES

Study	PWR Plant	Median Frequency	Recurrence Interval (Number of Reactor Years)	Reference
RSS	Surry	4×10^{-6}	250,000	6
IPPSS	Indian Point 2	4×10^{-8}	25,000,000	2
IPPSS	Indian Point 3	4×10^{-8}	25,000,000	2
ZPSS	Zion	3×10^{-8}	33,000,000	3
RSSMAP	Oconee	7×10^{-5}	14,000	7
RSSMAP	Sequoyah	5×10^{-6}	200,000	7
IREP	Crystal River-3	2×10^{-9}	500,000,000	8

maintenance procedures. The IPPSS accounted for testing and maintenance, including some procedures which are not in effect at all other plants.

2. Societal Risk

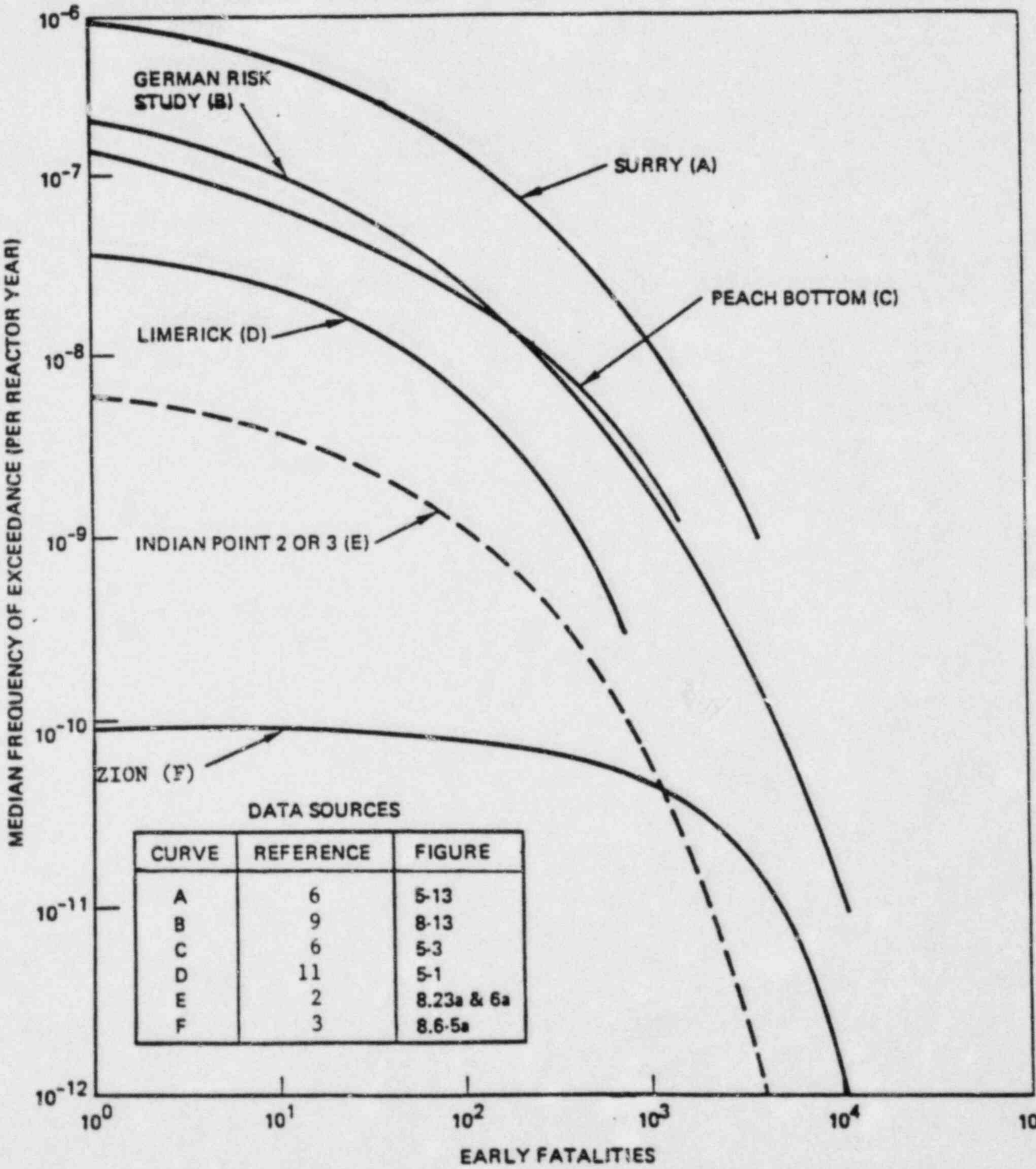
Societal risk comparisons have been compiled for the PRAs listed in Table 4. Graphical comparisons of results from these studies are presented in Figures 1 and 2. Many of the studies listed in Table 2 did not calculate risk curves and are, therefore, not included in Figures 1 and 2. The results from the German Biblis B risk study are included in these figures, as in the Staff table, because the study is recent, reasonably comprehensive, and analyzes a high population site.

Because so few PRA studies have comprehensively examined external initiating events as does IPPSS, the comparisons in these figures are for internal initiating events only. (The risk curves presented in the licensees' Question 1 testimony included both internal and external events.) Figure 1 shows the median risk curves for early fatalities as presented in the various studies, and Figure 2 presents similar results for latent cancer fatalities. These figures support the conclusion that Indian Point is within the range of societal risks posed by other nuclear power plants.

TABLE 4

PLANTS USED IN THE GRAPHICAL COMPARISONS

Plant	Reference
Indian Point 2	2
Indian Point 3	2
Surry	6
Peach Bottom	6
Zion	3
Biblis B	9
Big Rock Point	10
Limerick	11



CURVE	REFERENCE	FIGURE
A	6	5-13
B	9	8-13
C	6	5-3
D	11	5-1
E	2	8.23a & 6a
F	3	8.6-5a

Figure 1. Comparison of PRA Median Risk Curves for Early Fatalities (Internal Risk Only)

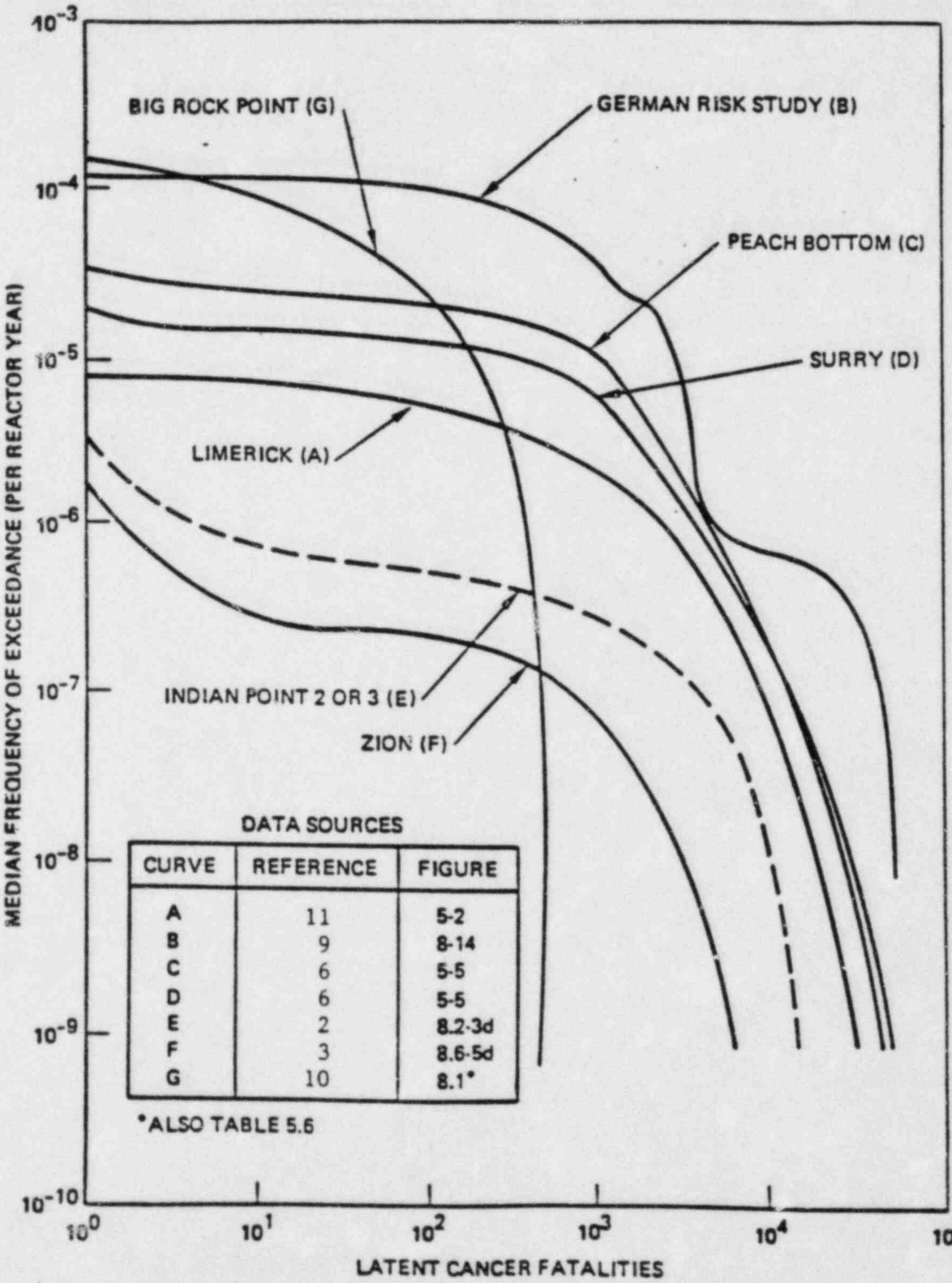


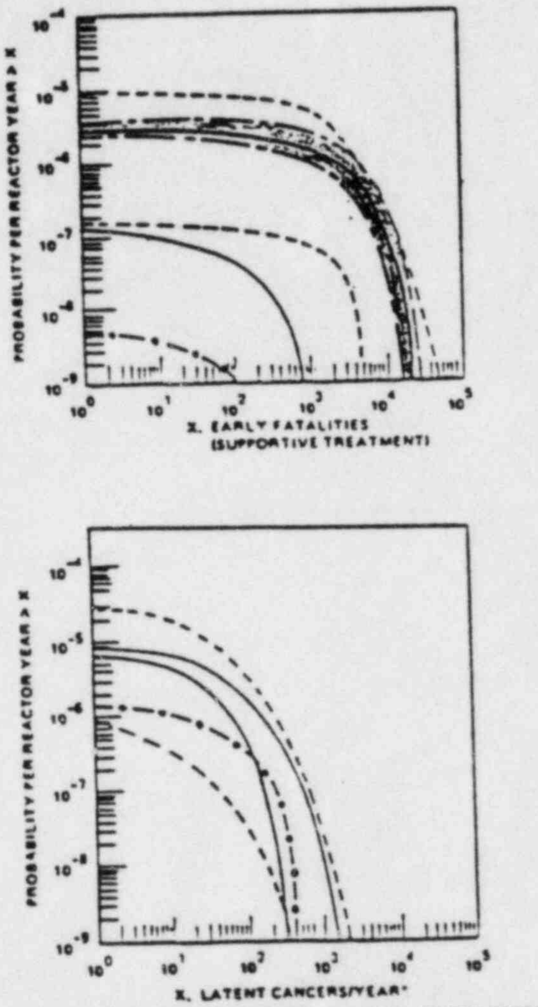
Figure 2. Comparison of PRA Median Risk Curves
for Latent Fatalities (Internal Risk Only)

B. Comparison with the Commission's Task Force Results

In 1980, a Commission task force studied the effects on risk of: (1) a typical pressurized water reactor (Reactor Safety Study, Surry) at different sites; (2) different plants at the same site (Indian Point); and (3) different public protection measures (Reference 4). Because the results of these studies are an indication of the range of risks posed by nuclear plants in general, they are also used for the comparison requested in Commission Question 5.

For this purpose, the median internal risk curves from the IPPSS are presented in Figure 3 along with results from Figure 11 of the Task Force Study for early and latent fatality risk. These curves support the view that the risk from Indian Point is within the range of risks from other nuclear power plants.

As can be seen from Figure 3, the early fatality risk curve calculated in the IPPSS lies more than an order of magnitude below the range of results presented in the Task Force Study. A large part of this difference results from the Task Force Study's failure to evaluate the strength of the containment, which precludes prompt containment failure. Thus, the Task Force did not include a release category for late containment failure. The IPPSS latent fatality risk curve lies within the range of the latent fatality risk calculated by the Task Force for the Indian Point site, and is below the range calculated for the Surry reactor at various sites.



NOTE 1. THE RANGES REPRESENT BEST ESTIMATES ON A COMPARATIVE BASIS. THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE.

2. PUBLIC PROTECTIVE MEASURES HAD NO SIGNIFICANT IMPACT ON TOTAL LATENT CANCER

ESTIMATED RANGE OF CONSEQUENCES FOR VARIOUS DESIGNS CONSIDERED AT INDIAN POINT SITE.*

ESTIMATED RANGE OF CONSEQUENCES FOR SIX SITES CONSIDERED WITH SURRY DESIGN.*

ESTIMATED RANGE OF CONSEQUENCES FOR VARIOUS PUBLIC PROTECTIVE MEASURES CONSIDERED AT INDIAN POINT SITE.*

INDIAN POINT 2 OR 3 — INTERNAL ONLY.**

*REFERENCE 1.
**REFERENCE 2.

Figure 3. Ranges of Risk Variation

V. SPECIAL DESIGN FEATURES AT INDIAN POINT

The comparisons presented in the previous sections indicate that the Indian Point Units are within the range of risks of other nuclear power plants, despite the demographic characteristics of the area surrounding the Indian Point plants. It is thus appropriate to ask whether this conclusion is supported by information about the engineering and design features of the plants.

Nuclear power plants located at high population sites have received special attention from regulatory agencies. During the licensing review for the Indian Point Units, additional features were incorporated into the plant designs to supplement the standard safety features. These features were highlighted in the Director's Order of February 11, 1980.

Among the features that could lead to lower frequencies of major releases from the Indian Point containments than from some other containments are:

(1) The design and construction of these containments, with a pressure limit of 141 psia and a large volume of 2.6×10^6 cu. ft., gives them the capability to withstand internal pressures well in excess of the design pressure of 62 psia. Additionally, the containments can withstand without significant structural damage all credible seismic events that could occur in this area. The containment building configuration allows gases to circulate and mix easily to prevent local accumulation of hydrogen. This configuration also provides for more effective containment heat

removal capability. In addition, the geometry of the reactor cavity promotes dispersion of the core debris, thereby increasing its coolability. Also, the geometry of the containment floor provides for easy entry of water to the reactor cavity to cool the debris.

(2) Containment cooling capability is provided by diverse systems. The design includes five fan cooling units in addition to four pumps capable of providing containment spray recirculation. The availability of any one of the fans or sprays is sufficient to prevent containment overpressure failure. Two recirculation pumps, located inside containment, are unique to Indian Point and are two of the pumps capable of providing containment spray.

(3) The Indian Point containments have two sumps that provide for recirculation of emergency core cooling water. The presence of two sumps is also unique to Indian Point.

(4) The presence of the recirculation pumps inside containment provides the capability of recirculating emergency core cooling water without its leaving the containment building.

(5) Three gas turbine-generators are available for supplying power to either unit. This feature is unique to Indian Point and provides an unusual degree of diversity in emergency power sources.

(6) Confirmatory signals (S signals) are sent upon actuation of emergency safeguards to certain power operated isolation valves to ensure that, if a valve had been inadvertently placed in an incorrect position, it would be restored to its correct position. This feature reduces the likelihood of bypassing the containment.

(7) The containment weld channel pressurization system and the isolation valve

seal water system help to assure that the containment leaktightness is maintained.

(8) The service water and component cooling water systems are arranged to maximize redundancy of active components. Any one of six service water pumps can supply any service water load. Similarly, either of two component cooling water pumps can be connected to any component cooling water load. The flexibility provided by these and similar interconnections within and between systems results in particularly low risk from internal initiating events at Indian Point.

The risk reductions afforded by some of the design features discussed above have been quantified using information from the IPPSS. For example, the frequency of late overpressure containment failure from internal initiating events is reduced by one to two orders of magnitude by the presence of fan coolers, which back up the spray recirculation system. The gas turbines, an additional source of AC power recovery for the time period of one to three hours following a core melt, provide up to an order of magnitude reduction in the frequency of late overpressure containment failures from internal initiating events. When external as well as internal initiating events are considered, the fan coolers provide up to an order of magnitude reduction and the gas turbines provide less than a factor of two reduction in the frequency of late overpressure containment failures. While not specifically quantified, the other design features discussed above would certainly provide further risk reduction.

On the strength of these special design features and other specific Indian Point systems, less than 2 percent of the internally initiated core melts lead to containment failure. As indicated in Table 2 and supported here, the frequency of a major release resulting from internal initiating events is thought to be less at Indian Point than at a number of other nuclear power plants. In addition, as stated above, the various safety features, particularly the fan coolers, provide significant reductions in overall (internal plus external) frequency of late containment failure.

As discussed in Licensees' Testimony of Thomas E. Potter on Commission Question Five (Mar. 22, 1983), the range of latent fatality risk among nuclear power plant sites, given a severe release, is relatively narrow. Based on the information in Table 2, the strength of the Indian Point containments, and the special design features at the plants, the release frequency at Indian Point is lower than the estimated release frequencies at many other plants. The narrow range of latent fatality risk, in conjunction with a lower than average release frequency, supports the conclusion that the Indian Point latent fatality risk is within the range of latent fatality risk of other nuclear power plants.

Based on the information in Table 2, the absolute value of the early fatality risk at a number of nuclear power plants is very low. At Indian Point, this is largely due to the strength of the containments, which essentially precludes prompt containment failure. The only accident contributing to early fatality risk is the interfacing systems LOCA which, as shown in Table 4, has a very low frequency of occurrence.

Special design features, together with standard nuclear power plant safety systems, result in very low early and latent fatality risk at Indian Point Units 2 and 3.

VI. CONCLUSIONS

Each of the several comparisons used in this testimony to address Commission Question Five supports the conclusion that Indian Point Units 2 and 3 are within the range of risks posed by other nuclear power plants.

A comparison of the Indian Point plants to the Commission's preliminary safety goals shows that these plants are within these goals. As such, they are in the class of plants whose risks are in a range below the limits established by these goals.

Various comparisons of the results of other PRAs to the results of the IPPSS show that the Indian Point plants are within the range of risks estimated for other nuclear power plants. Table 2 indicates that the early fatality risk for a number of nuclear power plants, including Indian Point, is very low. The Indian Point early fatality risk is low because, based on the strength of the containments, the low frequency interfacing systems LOCA is the only contributor to early fatality risk at Indian Point.

Using the source terms proposed in the previously submitted Question 1 testimony of Dr. William Stratton, Dr. Walton Rodger, and Thomas Potter, no early fatalities would occur for any Indian Point accident scenario.

When absolute risks are very low, differences between these low numbers are relatively unimportant.

With regard to the latent fatality risk, the Indian Point plants are close to the national average of the mean values of latent fatality consequences, based on the generic work reported in NUREG/CR-2239 (Reference 12). This report shows that the range of latent fatality consequences, given a specified release, is relatively narrow. See Licensees' Testimony of Thomas E. Potter on Commission Question Five (Mar. 22, 1983).

Based on the strength of the Indian Point containments and the special features of the Indian Point plants, radioactive releases from these plants would be less frequent than at many other plants. See Table 2. The narrow range of the consequences and the lower frequency of containment failure support the conclusion that the latent fatality risk is within the range of such risks posed by other nuclear power plants.

The above conclusion on latent fatalities is also relevant to the issue of whether any mitigation devices are warranted for the Indian Point plants. As discussed under Commission Questions 1 and 2, the principal application of these mitigation devices would be to reduce latent fatalities. The Indian Point plants have latent fatality risks which meet the Commission's preliminary safety goals and are within the range calculated for other nuclear power plants. This range itself will be lower and narrower with reductions in source terms. Therefore, no additional mitigation features are necessary to bring Indian Point within the range of risks posed by other nuclear power plants.

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NAME

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EDUCATION

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

PROFESSIONAL EXPERIENCE

General Summary

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

On USS Enterprise, Reactor Training Assistant, 5 months, 1971. Responsible for technical training of approximately 400 nuclear trained officers and men prior to annual safeguards examination. Propulsion Plant Station Officer, 9 months, 1970-1971. Responsible for maintenance and operation of one propulsion plant (two reactors, eight steam generators, and associated equipment) during power range testing of new reactors and during deployment. Approximately 80 enlisted personnel were assigned to the plant. Shift Propulsion Plant Watch Officer, 18 months, 1969-1970. Supervised a crew of about 20 navy enlisted operators and many shipyard workers on 8-hour shift rotation conducting maintenance and testing in one propulsion plant during refueling-overhaul. Shipboard qualifications: Propulsion Duty Officer, responsible for all propulsion equipment during absence of Reactor Officer and Engineer Officer. Engineering Officer of the Watch, operational watch in Central Control, responsible for all propulsion and engineering equipment and watch standers. Propulsion Plant Watch Officer, operational watch in one propulsion plant, directed and responsible for all operations in the plant.

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

Chronological Summary

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

MEMBERSHIPS, LICENSES, AND HONORS

The Society for Risk Assessment.
Institute of Electrical and Electronics Engineers.
American Nuclear Society.
American Association for the Advancement of Science.
The New York Academy of Sciences.
U.S. Naval Reserve, Commander.
Registered Nuclear Engineer, State of California.

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

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

D.F. Paddleford

Kansas State University, B.S. (1960) and M.S. (1962) in Nuclear Engineering

Post Graduate Courses in Engineering at UCLA, SMU, MIT

Mr. Paddleford is an Advisory Engineer in the Risk Assessment Section of the Nuclear Safety Department at the Westinghouse Nuclear Technology Division. Since joining Westinghouse PWR Systems Division in 1965, Mr. Paddleford has held a variety of positions in areas of increasing responsibility related to PWR plant safety, licensing, reliability, safety standards development, transient analysis, including core melt behavior, and probabilistic risk assessment. Most recently he has been engaged in the management and analysis of degraded core related issues, including test programs. He is currently active on AIF and IEEE Committees on Development of Risk Criteria and Utilization of PRA methods and is one of the principle authors of the Technical Writing Group which prepared the Industry/NRC "PRA Procedures Guides" under sponsorship of ANS and IEEE. He is a member of the IDCOR Technical Advisory Group and several IDCOR Expert Review Groups.

In the early 1970's his experience and responsibility included lead on research projects to develop a probabilistic approach to safety analysis, including systems reliability and data, probabilistic fracture mechanics models, core migration assessment, and probabilistic modeling of consequences associated with major fission product releases. Additional pertinent experience has included development of methods for parameter uncertainty propagation through design analysis computer codes and analysis of TMI and alternative scenarios at the request of the Kemeny Commission. Prior to joining Westinghouse, Mr. Paddleford was at Atomics International where he worked in areas of reactor physics and transient analysis in support of the SNAP 2/10 safety development.

Mr. Paddleford is a member of ANS and Sigma Xi and is a registered Professional Engineer. He is author or co-author of a number of papers on reactor safety and risk assessment.

NAME

THOMAS E. POTTER

EDUCATION

M.S., Environmental Science, University of Michigan, 1972.
B.S., Chemistry, University of Pittsburgh, 1963.

PROFESSIONAL EXPERIENCE

General Summary

Consultant on health and safety aspects of nuclear power. Performing environmental dose assessments for nuclear power plant safety analysis, environmental reports and operating reports. Assisting clients in design and implementation of radiological or environmental monitoring programs and interpretation of results. Providing independent review of in-plant radiological protection programs and effluent analysis programs.

Consultant in radiological health aspects of nuclear power. Prepared radiological health section of safety analysis reports and environmental monitoring programs and evaluated data from those programs. Developed a mathematical model to predict radiation doses from nuclear power plant effluents.

License administrator, plutonium fuel facility health and safety supervisor. Provided radiological safety review of major facility modifications. Used these analyses and nuclear criticality analyses perfomed by others to prepare AEC special nuclear materials and byproduct license applications. Served as corporate contact with AEC in matters related to licensing. Organized and supervised a radiological protection program for a plutonium fuels fabrication facility and hot cell facility. Instituted personnel monitoring programs using thermoluminescent dosimetry and breathing-zone areosol sampling in 1967. Served as secretary of a plant safety committee which inspected all operations and reviewed detailed written procedures for operators. Served as member of a corporate safety committee which determined corporate policy regarding health and safety matters.

Chronological Summary

1973-Present	Consultant, Pickard, Lowe and Garrick, Inc.
1972-1973	Consultant to Dr. G. Hoyt Whipple, University of Michigan.
1963-1970	Nuclear Materials and Equipment Corporation (NUMEC). License administrator, plutonium fuel facility health and safety supervisor.

MEMBERSHIPS

American Chemical Society.
American Nuclear Society.
Health Physics Society.
Certified by American Board of Health Physics.

REPORTS AND PUBLICATIONS

Woodard, K., and T. E. Potter, "Consideration of Source Term in Relation to Emergency Planning Requirements," presented to the Workshop of Technical Factors Relating Impacts from Reactor Releases to Emergency Planning, Bethesda, Maryland, January 12-13, 1982.

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Woodard, K., and T. E. Potter, "Probabilistic Prediction of X/Q for Routine Intermittant Gaseous Releases," Transactions of the American Nuclear Society, Vol. 26, June 1977.

Dennis C. Richardson - Risk Assessment Technology Manager

Penn State University, B.S. Aerospace Engineering

1963

M.S. Control Engineering

1965

San Diego State University, M.S. Mathematics

1970

University of Pittsburgh, MBA

1980

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

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

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

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POWER AUTHORITY OF THE STATE OF NEW YORK
10 COLUMBUS CIRCLE, NEW YORK, N.Y. 10019

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
4 IRVING PLACE, NEW YORK, N.Y. 10003

April 1, 1983

James P. Gleason, Chairman
Honorable Frederick J. Shon
Honorable Oscar H. Paris
Administrative Law Judges
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: In re Consolidated Edison Co. of New York, Inc. & Power Authority of the State of New York (Indian Point, Units 2 and 3), Nos. 50-247 SP, -286 SP

Dear Judges Gleason, Shon, and Paris:

Enclosed is an Errata Sheet for Licensees' Testimony of Dennis C. Bley, Donald F. Paddleford, Thomas E. Potter, and Dennis C. Richardson on Commission Question Five.

Sincerely,

Brent L. Brandenburg
Assistant General Counsel
Consolidated Edison Company of New York, Inc.

Paul F. Colarulli
Morgan Associates, Chartered
Counsel for the Power Authority of the State of New York

cc: Official Service List

Enclosure
PFC:BLB/pat

ERRATA SHEET

Page 6, line 13, vicinity should read "vicinity".

Page 8, line 3, "external events," should read "external events.".

Page 10, line 16, "Early fatality" should read "Early Fatal".

A revised page 11 is enclosed.

TABLE 2

RESULTS OF EXISTING PROBABILISTIC RISK ASSESSMENTS

*** WARNING - THERE ARE LARGE UNCERTAINTIES ASSOCIATED WITH THE VALUES PRESENTED IN THIS TABLE. ALSO, PRAs WERE NOT PERFORMED USING CONSISTENT METHODOLOGY AND ASSUMPTIONS

	PRA	NSSS/AE	DATE/POWER (MWe)		F CORE MELT [1] [2]	F MAJOR RELEASE [1] [3]	INDIVIDUAL RISK WITHIN 1 MILE: EARLY FATAL [1] [4]	CANCER FATAL [1] [5]	COMMENTS
ANO-1	IREP	B&W/BECHTEL	81	836	5×10^{-5}	2×10^{-5}	5×10^{-7}	2×10^{-7}	all-PWR 2 [3]
Biblis B	GERMAN RSS	FRG (W)	78	1300	4×10^{-5}	1×10^{-6}	3×10^{-9}	2×10^{-8}	Containment stronger and larger than U.S.
Big Rock [6]	WOOD-LEAVER/SAI	GE/BECHTEL	81	71	1×10^{-3}	0	0	—	Low power level, remote siting [9]
Browns Ferry	IREP	GE/TVA (BWR 4, M I)	81	1067	2×10^{-4}	4×10^{-5}	2×10^{-7}	1×10^{-6}	ATWS and interdependency in redundant RHR trains dominate core melt [10]
Calvert Cliffs	RSSMAP	CE/BECHTEL	82	850	2×10^{-3}	1×10^{-3}	9×10^{-6}	2×10^{-5}	More comprehensive IREP study in progress. AFWS redesign will lower risk and core melt frequency [11]
Crystal River	IREP	B&W/GILBERT	80	825	4×10^{-4}	2×10^{-6}	3×10^{-6}	2×10^{-6}	F core melt reduced by factor of 3 by procedure changes [12]
Grand Gulf	RSSMAP	GE/BECHTEL (BWR 6, M III)	81	1250	4×10^{-5}	4×10^{-5}	1×10^{-7}	1×10^{-7}	Containment always fails directly to atmosphere, does not assume staff's analysis of ATWS risk
I.P. #2 [6]	PLG	W/UE&C	82	873	4×10^{-4}	3×10^{-4}	3×10^{-8}	1×10^{-8}	Includes external events [13]
	PLG	W/UE&C	83	873	1×10^{-4}	4×10^{-6} a 5×10^{-5} b	7×10^{-9} c	6×10^{-8} c	Includes external events
	PLG	W/UE&C	83	873	3×10^{-5}	4×10^{-6} a 9×10^{-7} b	6×10^{-9} c	3×10^{-9} c	Internal events only
I.P. #3 [6]	PLG	W/UE&C	82	965	9×10^{-5}	3×10^{-6}	1×10^{-9}	3×10^{-10}	Includes external events [13]
	PLG	W/UE&C	83	965	5×10^{-5}	4×10^{-6} a 6×10^{-6} b	6×10^{-9} c	2×10^{-8} c	Includes external events
	PLG	W/UE&C	83	965	3×10^{-5}	4×10^{-6} a 5×10^{-7} b	6×10^{-9} c	3×10^{-9} c	Internal events only
Limerick [6]	SAI	GE/BECHTEL (BWR 4, M II)	81	1055	2×10^{-5}	3×10^{-6}	1×10^{-8}	1×10^{-6}	Mean value, assumes ATWS fix
Millstone	IREP	GE/EBASCO	82	652	3×10^{-4}	1×10^{-4}	1×10^{-7}	6×10^{-7}	Major release is in Release Category 2
Oconee	RSSMAP	B&W/BECHTEL	80	860	8×10^{-5}	4×10^{-5}	2×10^{-7}	1×10^{-7}	1/4-PWR 2; 3/4-PWR 3
Peach Bottom	WASH-1400	GE/BECHTEL (BWR 4, M I)	75	1065	3×10^{-5}	7×10^{-6}	4×10^{-8}	3×10^{-8}	Staff's analysis of ATWS would likely result in risk exceeding safety goal
Sequoyah	RSSMAP	W-IC/TVA	78	1148	6×10^{-5}	4×10^{-5}	1×10^{-6}	5×10^{-7}	H_2 control reduces risk by 2 to 3
Surry	WASH-1400	W/S&W	75	775	6×10^{-5}	1×10^{-5}	2×10^{-7}	1×10^{-7}	2/3-PWR 2; 1/3-PWR 3
Zion [6]	PLG	W/S&L	81	1100	4×10^{-5}	4×10^{-6}	2×10^{-8}	1×10^{-8}	Includes external events

1/ All numbers are median values or point estimates from internal initiators unless otherwise specified.

2/ Frequency of core melt 1×10^{-4} is the Safety Goal Value for Accident Probability Comparison.

3/ Frequency of release with Potential for Early Fatalities Assuming Nominal Evacuation and Warning Times (RSS).

4/ 5×10^{-7} is the Safety Goal for Early Fatality Risk Comparison. Same assumptions as 3 above unless specified.

5/ 2×10^{-6} is the Safety Goal for Cancer Fatality Risk Comparison. Same assumptions as 3 above unless specified.

6/ Utility-performed PRAs. All values are rough estimates based upon initial interpretation of results.

7/ Deterministic emergency response assumptions (1-hour delay with at least 3-hour warning) for dominant sequence when determining individual risk.

8/ Predicted risk is dominated by small LOCAs and transients. Source term reduction expected to reduce predicted risk to within guidelines. Likelihood of major release could be reduced by adding parallel valves at the discharge of the borated water storage tank or by improving DC power redundancy.

9/ Low power level (71 MWe) results in low individual risk. Extensive design modifications necessary to reduce core melt frequency.

10/ Reduction of core melt frequency would require redesign of the residual heat removal system to eliminate commonalities between trains which reduce the significance of multiple redundancy.

11/ AFWS redesign is expected to significantly reduce core melt frequency and individual risk. IREP study including improved AFWS design will be available in Spring 1983. Modification to DC power system and engineered safety system actuation system may be required to lower core melt frequency within guidelines. Predicted risk is dominated by transient event and should be significantly reduced by new source term data.

12/ Core melt frequency could be reduced to less than guidelines levels by improving written procedures and improving the reliability of the steam supply to the EFWS turbine driven pump. Predicted risk is dominated by small LOCA events. New source term information is expected to result in a moderate reduction in predicted risk.

13/ Core melt frequency is dominated by seismic considerations.

14/ Core melt frequency could be reduced to below guideline levels by redesigning the emergency AC power system to reduce dependency on the gas turbine and improving procedures for responding to transients. Predicted risk is dominated by transient events. New source term information should result in a significant reduction.

a. Frequency of release with potential for early fatalities assuming the nominal evacuation and warning times used in the IPPSS.
b. Frequency of release with potential for early fatalities assuming the nominal evacuation and warning times used in the RSS.
c. This is a mean value calculated using the IPPSS emergency response assumptions (1 hour delay with at least 8 hour warning

consequences or calculations involving the risk curve, specifically in section 3 the calculations dealing with the computation of individual risks and societal risk, 3A 3B, and section 4, part A-1, the calculations on individual risk, will usually be best handled by Mr. Potter.

Mr. Richardson and myself will kind of speak in general for the bulk of the testimony.

Q. Who wrote, on page 4, beginning with the second paragraph under section 3?

A. (Witness Bley) The entire testimony has been written by the group in general, with the assistance of others, so I can't name one specific person on this panel. But the panel will be glad to answer any questions on that paragraph.

Q. Who are the others who assisted?

A. (Witness Richardson) Myself, Mr. Paddleford.

A. (Witness Bley) People in our organizations. Dr. Kaplan, who has been here before, Dr. Bier on our staff. There were several others on our staff who have reviewed and commented, and his comments probably got into the testimony, and the attorneys for the utilities and their associates have reviewed and commented on it.

So it is really an effort of many people. Again, this panel speaks for the testimony and has essentially approved it all and will be glad to answer any questions about it.

JUDGE GLEASON: Mr. Bley, I thought you had indicated that Mr. Potter was responsible essentially for page 4, page 5 and page 6. Am I incorrect?

THE WITNESS: (Witness Bley) Your Honor, Mr. Potter is responsible for, as I said, the calculations with respect to individual risks and risk curves.

JUDGE GLEASON: I missed that. All right.

Q. Dr. Bley and Mr. Richardson, could you give us your understanding of what the preliminary safety goals represent in the sense of how were they arrived at numerically?

A. (Witness Bley) I can give you my opinions, and they are essentially opinions. The safety goals ca·e out of a fairly long process of discussions at the NRC staff, with information coming in from the AJRS, who he prepared a report on safety goals, the Atomic Industrial Forum prepared a report; other organizations commented.

Probably the original basis goes back to WASH 1400 which, in my own opinion, once it was published, stood as something like a de facto sort of goal because it was the only risk criteria that had been published.

In its final version with all of that background of information, looking at the studies that have been completed, where the risk was, and the final presentation of the goals they were stated in terms of a fraction, a tenth of a percent, of other existing risks.

So quantitatively that's the source, but it comes from a background of a lot of consideration about the risks of nuclear power plants as they were understood and other risks that society faced.

A. (Witness Richardson) Yes. I might add Westinghouse has been involved the last couple of years in our own discussions and also comments to the staff and also our own representation on some of the other groups, like the AIF, et cetera, on looking at both the need for safety goals and what form they should be in.

Of course, as we are all aware, there has been a variety of forms and formats and numbers generated the last couple of years. The present form that came out that Dr. Bley mentioned has been after -- at least a couple of drafts -- and we have always been involved in looking at these and shall be, over the trial period of these two years.

My personal opinion is that something like this is very important in terms of being able to put nuclear power, or anything else, in somewhat of a perspective related to risks that the society as a whole and individuals are under.

Q. Okay, I would like to ask a related question and I would like you to give me a yes or no answer, and you can elaborate if you need to. Although please stay fairly close to the question.

Are you saying that what the NRC did with the safety goals is to look at the calculated range of risk of plants and then to set the safety goals to correspond to what the various PRAs were showing?

A. (Witness Bley) That's not what I am saying, no. What I am saying is that the safety goals, and they include goals with respect to early fatalities, cancer fatalities, and core melt, were based on the principle of insuring that the contributions to the risk that the general public ses of nuclear power plants is small with respect to the other risks seen by the public.

On top of that I am sure the commission looked at the existing risks as calculated at a number of plants. We had in a cost benefit sense the value of the plants against the benefits of them, continuing and insuring that the risks to public health were low with respect to risks from other sources.

Then I suspect when it came to the core melt sort of goal they looked at plants that had risks that would have met the public health risk sort of goal and said what kind of core melt frequencies at those plants, consistent with meeting the health risk goals as the basis for coming up with a core melt goal, consistent with the previous health risk goals, which at least gives them a tool for looking at plants that haven't done a complete PRA out to the public health risk level.

Q. That's an interesting and plausible analysis, but you are not saying, are you, that the safety goals represent either an average risk of nuclear plants or a particular point somewhere in the spectrum of risks, are you?

A. (Witness Bley) I would say that the health risk goals -- and I say this not because they were selected that way but because of the comparisons of calculations that have been made against them -- I would say they tend to represent, as far as health risks, the upper end of the risks posed by the plants, and again the upper end of the risks that one would want to have -- would be willing to have these plants show if they are to contribute very little to the health risk of the general public.

Q. So that's what you mean by national benchmark, that it represents the upper end of the risks?

A. (Witness Bley) That's what I mean by that sort of a statement, that's right.

A. (Witness Richardson) If I may add, my own opinion on this as far as the health risk levels in their present form in the safety goal, I don't believe they were based in any way on results of any studies, at least that I know of.

I would believe they were more set in that way to try to insure that the risks imposed by nuclear power plants would be a very minor contribution to the overall risk that society and individuals live under.

A. (Witness Paddleford) I would like to add that they were established after some qualitative goals that were set up basically to, just what Dennis said, to make sure that the risks -- there were no undue risks presented to the memters of the public.

Q. Dr. Bley, could you tell us with regard to table 1, the table states that for core melt frequency, the two methods of the core melt frequency, and you have medium frequency, is that correct?

A. (Witness Bley) That's correct.

Q. What is it that you have for the first two entries, average early fatality, average, latent cancer.

A. (Witness Bley) Those are essentially a mean calculation from the risk curves themselves.

Q. Could you give us, as best you remember, the relevant figures for 90 percent confidence interval?

A. (Witness Bley) We have given those in our earlier testimony. I don't recall them offhand, but we have presented them before. They are somewhat higher than these numbers.

Q. In the event -- in the one case of core melt frequency, do you know whether that would be higher or lower than the figure expressed as the NRC goal of one times ten to the negative four?

A. (Witness Bley) Can you be more specific? Which of the numbers are you asking are they higher than the NRC goal?

Q. For Indian Point unit 2 and Indian Point unit 3, the 90 percent confidence interval in IPSS, is that higher or lower than one times ten to the negative four? Presumably it would have to be for Indian Point unit 2, would it not?

A. Internally plus external?

Q. Yes.

A. For Indian 2 it is certainly higher because the median is higher. For Indian Point unit 3 I don't specifically recall. It is close to the goal. It is in our previous testimony. It is in the amendment to the IPSS.

Q. Do you recall how much higher it is?

A. (Witness Bley) I don' recall if it is higher.

Q. For unit 2 do you recall how much

higher it is?

A. (Witness Bley) No, but we have given that previously and it is in the amendment.

Q. Would you know the 90 percent confidence interval for the earlier entries, average early fatality, average latent cancer for Indian point units 2 and 3?

A. (Witness Potter) I would like to respond to that quest n, partly to clarify a previous answer.

When Dr. Bley spoke of using the means to obtain the numbers for the average early fatality risk and average latent fatality risk, the means were used for the frequency of release, but for the rest of of the analysis; that is to say, the source term and the consequence analysis, point estimate values were used; that is to say WASH 14000, source terms, the way we have been using that term, and the S-1 consequence matrix assumptions.

Given that conglomerate of assumptions -- I can't specifically answer the question where the 90 percentile value would be, but, overall, the values here are higher than the mean.

I would guess they would approximate the 90 percentile value.

Q. Did you want to be more specific than that as opposed to just guessing?

A. (Witness Potter) I can't, not having performed the analysis, be more specific than that.

The numbers you see here are higher than the 50 percentile value and I would guess they would be approximating the 90 percentile value.

Q. Thank you.

A. (Witness Bley) Maybe I should add something to that.

Q. Go ahead.

A. (Witness Bley) For the first two rows in that table, the average early fatality rests within one mile and the average latent cancer risk is within 50 miles.

To do the calculation, we put that those would be mean average calculations to compare with the goal. That's the way we calculated them. We didn't calculate explicitly the whole range of uncertainty to be able to provide that.

Q. Thank you. With regard to table 2,

could you turn to page 11 of your testimony, which produces table 2, please.

There is also a corrected version. Could one of you read what's on the second line of this that begins with the word "warning."

A. (Witness Bley) "Warning: There are large uncertainties associated with the values presented in this table."

Q. Also?

A. "Also PRAs were not performed using consistent methodology and assumptions," which is what we also said in the text of our testimony.

Q. Do any of you smoke?

A. (Witness Bley) No.

A. (Witness Potter) No.

A. (Witness Richardson) No.

Q. Have you made any effort to quantify the uncertainties that are referred to in this table?

A. (Witness Bley) No. In fact we followed what I feel were the instructions of the commission and sta ement of question 5, looking at generic calculations of the range of risk, and we took this table without our entries as one statement of the range of risks from plants around the country as it was presented by staff, with no further calculations.

In fact, to do what you ask would require essentially a redoing of all of the studies here. We did not do that.

We did provide the full range of uncertainties for the Indian Point plan in our own previous testimony.

Q. What about the differences that were due to different methodologies being employed, did you attempt to in any way quantify or establish bounds on those sorts of differences?

A. (Witness Bley) We took this table exactly as it came from staff. We have made no modifications to it. Save our entries on Indian Point.

Q. You referred to the table 2 results as "a representative sampling of PWRs and BWRs, is that correct?

A. (Witness Richardson) I believe that's correct.

Q. Could you describe what kind of statistical sampling procedure you used to find that these were a representative sample?

A. (Witness Bley) Could you tell me

where we made that statement? I would like to look at the whole quote.

Q. Page 10, 9 lines up from the bottom. It reads, "This table generally reflects the range of risks from internal initiating events at United States nuclear power plants because it includes a representative sampling of PWRs and BWRs high population density sites."

A. (Witness Bley) Now that I look at it I put it in its proper context. This certainly is not meant to imply that this is a statistical representative random sampling sort of procedure. This list of reactors represents, in our opinion as we wrote this, a representative sampling of the types of power plants, the types of water reactors in operation in the United States at this time. It is not a statistical representative sample of risks from all of those plants. It is a broad mixture of the kinds of plants and the vintages of plants by vendor and specific design types.

Q. Now, by representative sample of the types of plants, you do not mean that plants are represented here in proportion to the number of those types of plants nationally, do you?

A. (Witness Bley) Not speci.ically. I

haven't checked to see if they are or they are not.
Q. What you mean is it is a good mix of plants for showing different types of reactors, is that correct?
A. (Witness Bley) I think that's fair.
Q. You recall your earlier testimony during question 1 about how PRAs can come up with lower and more realistic risk estimates by eliminating gross conservatisms, do you not?
A. (Witness Bley) I do.
Q. And in general it is your opinion that IPSS is a relatively sophisticated and thorough PRA in this regard, is it not?
A. (Witness Bley) I agree with that.
Q. Are there other PRAs that you would see -- that you would place as being as thorough and realistic as IPSS in this regard?
A. (Witness Bley) The only one on this list, and there are some, I think, that are now about to come out or are coming out that maybe would be a design study, but I hasten to point out that the context of those remarks about additional care and the analysis narrowing the uncertainties has to do with an analysis that purports to be a complete one.

The other analyses on this list have left out what can be an exceedingly -- left out on purpose, intentionally -- a category of events that we found to be fairly important, and that's the external events. No compensation for leaving those events out was made.

The results weren't stretched to account for the events not studied.

So if those studies were carried out more thoroughly to pick up the external events, that could only add to the risk that you see at those plants. It wouldn't make it lower.

Q. I understand your point about the need to compare internal events with i.ternal events, if that's what the earlier studies only showed.

However, it is your testimony that with regard to the degree of sophistication and eliminating gross conservatisms for internal events, Zion is the one study that's comparable to IPSS; that is what is you stated, is it not?

A. (Witness Bley) No. That was a statement to cover the totality of a PRA. With respect to internal events only, I suspect some of the others would fit that category, too, although

I have not studied these other PRAs in depth.

Q. Are you aware of any that would be less sophisticated and realistic than IPSS?

A. (Witness Bley) I haven't studied them in great detail and I would be hesitant to characterize them one way or the other. Some of them I am sure are very complete, others probably are not.

Q. By "complete" you mean they progress far toward eliminating gross conservatisms?

A. Some probably do, yes.

Q. And some probably do not?

A. Yes. I am not willing to speak to that point because I have not studied those in great detail. Any of them, really.

Q. Do you believe that a knowledge of the extent to which different PRAs do that would be relevant for drawing comparisons with IPSS?

A. It would certainly be relevant. As of this point in time it is not directly possible, and the kinds of comparisons we have given here to search for available indications of the range of risks, knowing they are not fully comparable, I think is a reasonable approach. That's why we took more than one approach.

We took these various studies. We took the commission task force results and used those as another, if you will, surrogate expression, range of risks, and we took the safety goals as something of an upper limit sort of thing, although not a strict limit; an upper end of the range estimate on the range of risks.

So all of these give us things to compare with. Although none of them are perfect, the total is, in every case, supportive of the conclusions we draw in our testimony.

Q. So the core melt frequency for Indian Point unit 2 is right at the upper end of the range of the core melt conception of risk, is it not?

A. (Witness Bley) No. If you will go back to my statement of what we felt the goals -- what I felt the goals meant for the health risks, they compare with the upper end of the range of risk. The core melt goal in my opinion has come about, in my opinion, as a reflection for the plants that sort of meet the health risk goals, what sort of core melt for those would have been consistent.

If we look down the list of core melt

frequencies on table 2, we find several that are quite a bit higher than Indian Point.

Q. But it is true that core melt frequency or internal plus external events, the median frequency for Indian Point 2 is exactly equal to the NRC goal, according to your table 1; that's a correct, is it not?

A. My Bley after it is rounded off, yes.

Q. By the way, your table 2, is that taken from inside the NRC?

A. I think we cite this reference.

It is reference 5 to our testimony, a memo from William J. Dirks to the commission, draft dated January 5, 1983, as an attachment.

Q. I had heard that the layout of this table was somewhat different than the one in the Dirks memorandum and it corresponded to one inside the NRC. It may well be the substance of the two tables is the same, but I wonder if you checked.

A. (Witness Davidoff) (Witness Richardson) I don't recall.

A. (Witness Bley) I don't recall.

Q. With regard to this data in table 2, it is rot exactly the same as the numbers generated by the PRAs themselves, is it?

A. (Witness Bley) That's true.

Q. It involves some recalculation done by members of the NRC staff?

A. (Witness Bley) That's true.

Q. But you don't know how those recalculation is were done, do you?

A. (Witness Richardson) No.

A. (Witness Bley) No.

Q. Turning now to your table 3, this comparison does not really represent all these various plants as they currently exist, does it?

A. (Witness Bley) I would agree with your statement because, to my knowledge, the Surry plant and the safety study has adopted maintenance practices which would reduce their frequency.

My understanding is that the RSSMAP studies left out, at least to some extent, the improvements due to maintenance at the specific plants.

So I think the other plants, at least some of them would have lower frequencies of these events than are presented in this table.

Q. Are you aware how much lower?

A. (Witness Bley) No. I am not aware of

the specific maintenance practices at any of the plants on this list, other than the Indian Point plants.

Q. Do you have any kind of bounding estimate on how much lower it would be?

A. (Witness Bley) I can draw some inferences. From the reactor safety study design there is essentially a two-check valve discharge system which, on a bounding level, wouldn't be as reliable as the three valve sort of arrangement we see at Indian Point.

I don't know the other plants. So that it probably isn't as low as Indian Point but it may well approach it.

Q. So, in general, for the Oconee, Sequoyah and Surry plants, it is true that there are some sorts of improvements that are not reflected in this table?

A. (Witness Bley) I know it is true for Surry . My understanding is it is probably true for Oconee and Sequoyah, but I am not certain of that.

Q. Does this table include improvements that have been made for IPSS -- I am sorry, for Indian Point?

A. (Witness Bley) It includes Indian Point as it exists today, yes.

Q. Does that include the improvements that were made in reference to the IPSS amendment one?

A. (Witness Bley) My memory is we didn't reference this in reference one. If we did, the answer would be yes. I don't think it showed up in amendment one.

It would represent the discussion that was presented in our question one testimony and the results that were presented there.

Q. So the recent fix with regard to hurricanes and fires, and so forth?

A. They don't affect the C scenario. The only thing that would affect the V scenario would be the testing program on both the test valves and the RH gate valves, which I think we described earlier.

Q. The other places where you do have more general figures for Indian Point, such as, for example, in table one, those do include the fire and hurricane fixes, do they not?

A. (Witness Bley) That's true. All the results given as related to IPSS with respect to

Indian Point in this testimony include the fixes that we have discussed earlier, yes.

Q. So it would be fair to say that in general you are more familiar and conversant with the recent improvements with Indian Point than with various other plants?

A. (Witness Bley) In all aspects we are more familiar were with Indian Point than other plants. We have studied Indian Point in detail. We have not done that for the other plants.

Q. On page 20 of your testimony -- this is now the beginning of the third paragraph, "Features that could lead to lower frequencies of major releases from Indian Point containment than from some other containment."

The rather vague phrasing is intentional, is it not? When you say "could lead to lower frequencies"?

A. (Witness Paddleford) Yes.

Q. And, in general, you have not, for most of these eight at least, you have not calculated specific amounts of risk reduction attributable to them?

A. (Witness Richardson) We didn't do anything specific here, but we certainly know there are containment of lower volume and lower fire pressure. Anything in that direction would obviously be about the same or higher in terms of their containment capability and similar types of accidents.

A. (Witness Bley) Beyond that, we studied Indian Point as it exists. We found dominant contributors, and we studied several potential changes to the plant to see what improvements could be made in the risk.

We made no determined effort to see how much worse the plant could be made by deleting equipment and functions that are there and are intended to stay there.

So we have not, quantitatively, in detail, evaluated the worth of these separate items which we provide here, which, in total, we are sure will have an important effect, but we have not evaluated that singly or in total from the aspect of plant risk.

Q. Mr. Richardson, you stated that you are aware that there are some plants that have smaller containment and lower failure pressures, did I hear you correctly

Q. (Witness Richardson) Yes.

Q. You are also aware, are you not, that there are some plants that have higher failure pressures?

A. (Witness Richardson) Yes.

Q. Are you aware of any that have larger containment?

A. (Witness Paddleford) Yes.

Q. How many plants are you aware of that have higher failure pressure than Indian Point?

A. (Witness Richardson) The only one I know for sure would be the Seabrook containment because that has a higher seismic design basis, and I believe has a slightly higher onset of yield.

Q. Have you attempted to rank the failure pressures of containment of nuclear plants generally?

A. (Witness Richardson) No.

A. (Witness Bley) No.

Q. With regard to various equipment or containment cooling capacity, are you aware of the study NUREG CR 2069, put out by Oak Ridge National Laboratory, entitled, "Summary Report on a survey of lightwater reactor safety systems"? Is any member of the panel aware of that study?

A. (Witness Paddleford) No.

Q. Well, perhaps if I identified it a little further. It is authored by Fred A. Hedelson and contains a fairly systematic comparison of safety systems of a large number of nuclear plants, including the two at Indian Point.

Does that refresh your recollection that there is such a study?

A. (Witness Bley) Not mine.

A. (Witness Richardson) No, I have never read such a study.

Q. Thank you. On page 22 of your testimony, you state that the "frequency of laid over pressure containment failure from internal initiating events is reduced by one to two orders of magnitude by the presence of fan coolers which back up the spray and recirculation system."

Do you see that?

A. (Witness Paddleford) Yes.

Q. Now, that's true only for internal initiating events, is it not?

A. (Witness Paddleford) That particular statement is true for only internal initiating events.

However, later in the paragraph of the statement for internal plus external is given as up to an order of magnitude.

Q. Well, focusing on your first question about the fan coolers, it is true, is it not, that fire is classified as an external initiating event?

A. (Witness Paddleford) Yes.

Q. As, of course, seismic events would be, correct?

A. (Witness Paddleford) That's true.

Q. So the major sources of common mode failure that are likely to result in a loss of electrical power are not included in internal initiating events, are they?

A. (Witness Bley) Read the question back,

(The reporter read the last question.)

A. (Witness Bley) That's a little hard to answer directly, especially for me at this time.

The fire events generally don't totally lose electric power. They tend to lose key pieces of electric power. With regard to key pieces of electric power I think I would agree with your statement.

With regard to complete loss of electric power, I suspect -- I just don't know. I can't answer that. I don't remember well enough.

Q. It is true, is it not, that the Biblis B. Reactor was not licensed by the Nuclear Regulatory Commission?

A. (Witness Bley) That's true.

Q. Where is the Biblis B reactor located?

A. (Witness Bley) It is in Germany. I can't give you the exact location.

Q. You would acknowledge, would you not, that there may be some special safety features that certain other nuclear plants have that the Indian Point plants do not have?

A. (Witness Richardson) When you say "safety features," other plants play, because they are designed, have components that they have designated as safety features and are required for that plant in terms of safety. Indian Point plants may not have that equipment but they may not need that equipment.

So, when you say "safety features," that covers a broad line of equipment.

A. (Witness Bley) I can't think of specific features right off the top of my head now, at other plants that would have significant impact on risk at Indian Point, if they were at Indian Point, that Indian Point doesn't have.

That doesn't mean there can't be such things but none come to my mind.

Q. You will have to forgive me if this question is the same one you just answered. My colleague feels it is a different question. Are there any design features that Indian Point does not have that other plants do have that could contribute significantly to risk?

A. (Witness Bley) At Indian Point?

Q. Yes. To reducing risk.

A. (Witness Bley) I can't think of any that would significantly reduce the risk at Indian Point.

If you will suggest some, we could comment on them.

A. (Witness Richardson) I can't either.

MR. BLUM: We have no further questions.

JUDGE GLEASON: Ms. Moore.

CROSS-EXAMINATION

BY MS. MOORE:

Q. Good afternoon, gentlemen. I just have a few questions.

In the safe goal calculations on pages 46 of your testimony, I believe you said under cross-examination that the S-1 consequence matrix assumptions were used, is that correct?

A. (Witness Potter) That's correct.

Q. Could you refresh my memory a little bit and tell me what emergency response assumption the S-1 matrix includes?

A. (Witness Potter) Each of the four S matrix elements uses the same emergency response assumptions. The only difference between each of the four was essentially scaling of the dose by a factor of two, for the S-2 matrix, a factor of one for the S-1 matrix, a factor of .5 for the S-3 and .1 for the S-4 matrix.

The emergency assumptions involve evacuation of population within ten miles, sheltering of the population -- of 90 percent of the population from ten to fifty miles; normal activities for the rest of the population beyond ten miles; and for the entire population beyond ten miles, it is assumed ground dose exposure period would be 24 hours.

Within ten miles the important parameters involve delay of the evacuees. We handled that probable his particularly. It is fairly complex, but in general in 90 percent of the weather scenarios population was delayed -- elements of the population were delayed from half an hour to two hours; in 7 percent of the scenarios elements of the population were delayed from one and one half hours to three hours; and in three percent of the scenarios elements of the population were delayed from two and a half hours to four hours.

Then there was in a weekday school in session scenario, there was a special population group that had much longer delay times.

That's a summary.

Q. And if I remember correctly, that response, that emergency response scenario does not differentiate between internal and external events, is that correct?

A. (Witness Potter) That's right. It was applied to all much these categories.

Q. That S matrix includes also, does it not, the U factor source term reduction?

A. (Witness Potter) The entire matrix does, but the safety goal estimates do not.

Q. They do not, thank you.

I would like to turn your attention now to page 20 to 22 of your testimony. Could you please describe how your lists of design features contained on these pages was derived?

A. (Witness Paddleford) Combined from some things that we knew, special features that were included in the Indian points when they were first licensed, along with some other features that have been in the plant since the beginning but are not common to a very large percentage of nuclear power plants.

Q. On page 20 of your testimony at the bottom, going to page 21, could you give me the basis for your statement that the Indian Point units have more effective containment heat removal capability?

A. (Witness Paddleford) Which statement was that again?

Q. At the bottom of page 20, going over to page 21.

A. (Witness Paddleford) This is in regard to the more open containment structure which permits natural circulation, and the ability to get heat from all over the containment to the fan coolers or containment spray, which would not be possible if you had a highly compartmented situation.

Q. Could you tell me how the Indian Point containment in its ability for heat removal capability compared to, say, Crystal River?

A. (Witness Paddleford) I don't really know that much about Crystal River.

Q. Do you know about any of the other plants -- could you compare it to any of the other plants on the list that you had in table 2?

A. (Witness Paddleford) The Surry plant I know very well. Some of the differences are the cavity geometry underneath the reactor vessel is open, it is a low spot in the containment. It is an area where water would collect.

This wasn't true of the Surry-type design. That's one example.

Q. Are you aware whether other pressurized water reactors with large dry containment have diverse fan coolers and containment sprays?

A. (Witness Paddleford) Yes, there are others that do and others that don't.

Q. Do you know how many do?

A. (Witness Paddleford) I have not made up a list.

Q. Have you got an estimate of how many

might?

A. (Witness Paddleford) My guess is that a substantial number don't, like on the order of 50 percent.

MR. BLUM: Your Honor, I would object to a guess being given. I think that's too speculative to go into the record.

JUDGE GLEASON: I guess he can't do any better than that.

Can you?

THE WITNESS: (Witness Paddleford) No.

MR. BLUM: Then the answer should be that he just doesn't know.

JUDGE GLEASON: He didn't say he didn't know. He says he guessed about 50 or thought about 50.

Can you make it any more accurate than that?

THE WITNESS: (Witness Paddleford) On a number basis I can't give you more than I have given you, except that many plants have only a spray system and don't have accident fan coolers.

Q. I am correct, am I not, that you stated previously you haven't analyzed the risk reduction potential of these design features, is that correct?

A. (Witness Paddleford) In general that's true, but a couple of them we have analyzed them. The risk reduction significance.

Q. Would one of those features be the risk reduction potential from having two recirculation pumps inside containment?

A. (Witness Paddleford) No, we did not.

Q. However, was the existence of these two pumps taken into account in the PRA analysis itself?

A. (Witness Paddleford) Yes. We took credit for the two recirculation systems in the IPSS study.

Q. And the fan coolers were also taken into account, were they not?

A. (Witness Paddleford) That's correct.

Q. What about the S signals? Were they taken into account in the PRA?

A. (Witness Bley) We talked about this, I think, under question 1. I am not sure if it showed up specifically in the documentation of IPSS. It was considered and it was included in the evaluation of how likely it would be that the containment would remain unisolated if the venting were in process.

So it was considered, we discussed it in our testimony. I am not sure if it actually shows up in the IPSS documentation itself, quantitatively.

Q. And was the containment weld channel pressurization system considered in the PRA?

A. (Witness Bley) Only in the same respect that I just mentioned. I don't think it is specifically there but it was considered and weighed in the evaluation of possible containment bypass.

Q. Arc you aware whether the Zion facility, for example, has a containment cooling feature which Indian Point does not have?

A. (Witness Paddleford) Yes. The Zion plant does have an additional spray pump.

Q. Is it independent of the AC system?

A. (Witness Paddleford) No. From the standpoint of -- it has its own diesel as far as driving the spray pump but it is not independent from the the electrical system through a cooling mode.

A. (Witness Bley) Two reasons. It needs AC power for cooling and AC power to operate valves in the system.

Q. In your discussion of the service water system in the component cooling water, is the interconnectiveness of the service system achieved by the use of a common header?

A. (Witness Bley) Essentially, yes, it is possible to arrange any of the pumps on to any headers. There are separate headers in the case of service water, but they can be cross connected or isolated. It is possible to align them in any configuration.

Q. And what about the component cooling water system?

A. (Witness Bley) It can be arranged so any pump can provide any cooling through a common header arrangement, yes.

Q. Was the existence of this common header, the disability taken into account in the PRA?

A. (Witness Bley) Yes.

Q. Was it determined whether the existence of such a common header would render the system susceptible to failure from one pipe break?

A. (Witness Bley) I am afraid this moves into an area that is in our continuing work. It was considered and further work, especially on the component cooling system, has shown that while there are several places where a large break could disable the system, the chance of that happening is extremely remote because of the quality piping system, the pressures involved, the make-up capability of the systems and the energy required to actually puncture or shear one of the major pipe lines.

So we did consider it originally more recently, and it is not in the work that's been submitted to the board, it is not fully completed. We have looked even further the further work supports our view that such breaks are extremely unlikely.

Q. I believe you state in your testimony that one fan cooler is sufficient to maintain containment cooling, is that correct?

A. (Witness Paddleford) That's correct.

Q. Is the basis for this statement documented in the PRA?

A. (Witness Richardson) I am not sure if it is specifically in the PRA but the basis of it is that one cmponent is adequate for removal of the decay heat.

JUDGE GLEASON: I didn't hear the response. I didn't hear the second part.

THE WITNESS: (Witness Richardson) This was supported by work that we at Westinghouse had done after the submittal of the PRA, and the basis for it is that one component is adequate for removal of the decay heat.

MS. MOORE: I have no further questions, your Honor.

JUDGE GLEASON: Are you going to have redirect, Mr. Brandenburg? How much redirect do you have?

MR. BRANDENBURG: Just a couple of questions.

JUDGE GLEASON: All right. I am trying to take a recess, but let's go ahead.

REDIRECT EXAMINATION

BY MR. BRANDENBURG:

Q. Now, on cross-examination Mr. Blum asked you to identify plants which had either a smaller volume -- excuse me, a larger volume containment or a higher failure pressure.

I would like to ask you in a more global sense if you are aware of plants which have containment with either lesser volumes or lower failure pressures by general design category, and things of that sort?

A. (Witness Paddleford) Yes. In PWRs the ice condensers all have smaller volume and smaller pressure.

In BWRs, all the Mark 1s and Mark 2s have smaller volume. Some of those have higher pressure, but across the board I think they have smaller volumes.

MR. BRANDENBURG: That's the only question I have, Mr. Chairman.

MR. COLARULLI: No redirect, your Honor.

JUDGE GLEASON: Gentlemen, you are excused. Thank you very much.

MR. COLARULLI: Your Honor, just one matter. This is the last time that Mr. Richardson will be --

JUDGE GLEASON: Is this a correction of testimony, or what?

MR. COLARULLI: This is a question that Judge Shon raised during the questioning to the testimony. He raised it on 6449 of the transcript.

At that point in the proceeding there

was a discussion of steam generators and two ruptures and the further work that had been done on it. The witnesses had begun talking about guillotine type breaks for a tube.

Judge Shon, on 6449, asked a question concerning a situation in which there was a guillotine type rupture of a major pipe in the system and a steam generated tube rupture at the same time. He asked both Mr. Richardson and myself to check into that to see if any analysis had been done.

Mr. Richardson has a brief response to that question.

JUDGE GLEASON: Is that the transcript reference, 6449?

MR. COLARULLI: Yes.

JUDGE GLEASON: Go ahead, Mr. Richardson.

THE WITNESS: (Witness Richardson) Westinghouse has in the past performed analyses of a postulated event consisting of a simultaneous steam generator tube rupture and a large loss-of-coolant accident. Postulating such an event constitutes a double failure. As such, the ECCS acceptance criteria embodied in 10CFR50.46

and the requirements of the Appendix K model, with the inherent conservatisms, can mask the expected response of the reactor coolant system and core to postulated tube leakage during a large LOCA. Therefore, conservative better estimate calculations are performed to determine the potential impact of a secondary to primary steam generator tube leakage.

The design criteria for the steam generator tubes, the existing inspection program and the tube plugging criteria are designed to guard against the probability of tube ruptures in the event of a LOCA.

These better estimate calculations of a postulated large break LOCA have been performed with and without secondary to primary steam generator tube leakage, for a typical W plant. The calculation was performed for a double-ended cold leg guillotine break. The result of that calculation was that less than a 10 degree increase in the peak fuel cladding temperature occurred when a 250 gpm secondary to primary leak was modeled. This leak rate is consistent with a postulated do you believe-ended break of a steam generator tube. This calculated increase in peak clad temperature was due to the increase in steam binding in the reactor coolant loops.

Steam binding is a phrase denoting the resistance to venting steam generated in the core to the break. Increasing steam binding retards the reflooding of the core by the ECCS following a large break LOCA.

The ten degree change in peak cladding temperature for a single double-ended steam generator tube rupture combined with the calculated peak clad temperature for the case studied of 1567 F would indicate that a very large number of tubes would need to have double-ended ruptures before the peak clad temperatures would approach the limit of 10CFR50, Appendix K.

If a plant specific analysis were to be done for Indian Point, using the assumptions outlined above, the calculated effect would be comparable.

JUDGE SHON: Thank you. That's exactly what I wanted to find out. There would have to be many tubes ruptured before you would exceed Appendix K.

THE WITNESS: (Witness Richardson) Yes.

JUDGE SHON: Thank you.

(There was a pause in the proceeding.)

JUDGE GLEASON: We will start with Mr. Shelly tomorrow morning.

We could take a couple of minutes to argue, if you want to, Ms. Fleisher, your motion on excluding the testimony that came in the past week from the two counties. Do you want to argue that now?

Do you want it decided now? We will let the other parties argue it.

MS. FLEISHER: I understand. It would be hard for me to argue it any further than the paper I turned in without understanding the circumstances around the request of the subpoena. It would appear that the subpoena doesn't make it clear to you that they have not otherwise attempted to reach these people and ask them when they testified.

That's what they testified when we asked them about it. It seems to me, therefore, that issuing a subpoena was only used as a device so they could question these people and that is unfair, misleading, and a whole bunch of other adjectives.

JUDGE GLEASON: I didn't really ask for you to argue it again -- I didn't intend to suggest that because your motion was fairly complete. I just wanted to find out whether you concurred in hearing arguments on it and getting some decision. That's all.

MS. FLEISHER: I asked MR. Lewis if there was anything in the way of notes on the form which one signs. I remember when I got the subpoena for Mr. Fisher, Judge Carter put me through quite a grilling as to why I wanted it. Perhaps even though there is no such thing now, one should put on that form some request or some note that states that you have otherwise tried to reach the person.

JUDGE GLEASON: Thank you for your suggestions, Ms. Fleisher.

Mr. Levin.

MR. LEVIN: We are prepared to argue Ms. Fleisher's motion and obviously we oppose it.

First of all I would like to point the board to page 11125 of the transcript. That date was March 23, 1983.

At that point the Chairman was speaking, and this goes to the question of whether or not there was notice of the appearance of these witnesses in advance of their appearance. At that page, I believe it is the Chairman speaking, saying, "Thirdly, I would just note for the record a request for a subpoena which was made by the Power Authority to have the appearance in this proceeding of Mr. Philip Schmer from the Orange County National Disaster and Civil Defense and Michael Scalpi, Office of Civil Defense, Putnam County. I just wanted to note that."

"MR. CZAJA: Maybe we should note for the record that the subpoenas will be returned 2 o'clock p.m. on March 30 and we will anticipate on that date that we will be able to put Messrs. Scalpi and Schmer on the stand at that time."

So as to whether there was official notice on the record as to the apparance of those witnesses, there most certainly was.

I do not know whether Ms. Fleisher was present at that time, although the --

MS. FLEISHER: I tried searching the notice, too. It doesn't seem to me there is anything in the CFR that talks about the notice of a subpoena.

MR. LEVIN: Your Honor, if you would

let me finish my argument.

JUDGE GLEASON: Let him finish, Ms. Fleisher.

MR. LEVIN: There was notice on the record, point one.

Point two, Ms. Fleisher claims that -- I believe this is at page 2 of her motion, "I asked if they would have testified without a subpoena; they said yes." Then she cites the transcript at page 12181.

Looking at 12181 of the transcript I find absolutely no such statement. I find a question by Ms. Fleisher and I quote, "Mr. Schmer, why was it necessary for you to have a subpoena to appear here today?

"A. I couldn't answer that question. I have no idea.

"Q. Had you been asked to come to testify voluntarily?

"A. Other than that subpoena, no."

So there is no concession that had the witness been asked to come voluntarily that he would.

Point three, these are public officials, not under the control of the Power

Authority of either licensee. The only assurance that we had, regardless of what they might affirm to us in person, the only assurance we had that they would be here was by way of a subpoena.

I further note that is not a novel process for this or any other proceeding. New York City Council, for example, called by subpoena someone they arguably would have had much more control over, Inspector Littlejohn. He appeared by subpoena, gave direct testimony, just as did these witnesses, and was cross-examined.

Having said all that and arguing that there is absolutely no basis to strike this testimony, I would like for the board to know that the Power Authority has no objection whatsoever to these witnesses being recalled for whatever further cross-examination Ms. Fleisher, or any other intervenor feels is necessary. Although, with the caveat, that we would not concede that any of that time should come from the time that we have allotted for cross-examination during the final week of emergency planning testimony.

MS. FLEISHER: Your Honor, we have called many public officials and they have come. I think that to state that a public official can't be depended to come and that a subpoena is pretty much of a strong statement forcing them to come.

I was not present. If I was present I was out of the room at the time of his quotation.

Now, I understand that we can't go around chasing each other and giving notice. There has been a great laxity of notice at times in these hearings. I feel there are plenty of people who don't know what the schedule is this week because they didn't happen to be here late last Thursday.

It is hard on those of us who haven't the forces and haven't the great dough to provide people sitting here all the time. This hearing was originally started by the Union of Concerned Scientists' application. All of you must have known that you were going to have to deal with a bunch of hams that don't have any money and don't have the privilege of having somebody here to watch for such notice.

If Mr. Levin was so concerned about getting these people here and all, he might have let us know. I know that Mr. Schmer's name would have electrified me if I had heard it because I know him and I have spoken to him before and I would have wished to be more prepared.

Evidently I was in possession of the Orange County plan which the state had sent to all of us, but I didn't think to bring it that day.

You can see that I come pretty well loaded sometimes. If I would have known about him, I would have prepared for it him.

JUDGE GLEASON: I regret very much is that you were surprised by the appearance of those witnesses here. I did bring to the attention of those in attendance that an application for a subpoena had been made. There is an obligation on the part of all parties to be present during these proceedings. Under the rules the chairman and the board does not have much discretion with respect to the issuance of subpoenas.

As far as the relevance of the testimony, I don't think there is any question in our minds that that testimony or the testimony from those two counties was at least relevant in these proceedings. At least some of us had asked in our own minds off the record as to why there was not somebody here from those two counties since parts of those counties are in the EPZ.

Let me just say if there is any time

in the schedule, and I can not assure that because, as you know, we are on a very tight schedule, we will try to provide the these witnesses for further cross-examination. But I can't give you that assurance.

MS. FLEISHER: I feel that the past record is a mess and I really resent that we should have to sit on that rec›rd when we were so unaware of what was happening.

I don't think that we had a proper explanation yet of why they resorted to subpoena. And also why it was even beholden to Con Ed and the Power Authority to bring these people in. If these people were such an important part of the general picture, they should have been invited by the board.

I mind this very much. I don't feel a bit satisfied by what has been said by Con Ed and PASNY people.

JUDGE GLEASON: I don't know whether anything that I would say would satisfy you, Ms. Fleisher. All I can say is what it is in my mind to say, and I say that you have an obligation to be here and we have tried to be fair to all parties, and and I think we have been. And if the record is deficient, it has certainly not been because there hasn't been an allocation of time granted by this board to the intervenors.

The motion is denied and we stand in recess until 9 o'clock tomorrow morning.

(Hearing recessed at 4:40 p.m.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before

THE ATOMIC SAFETY AND LICENSING BOARD

in the matter of: CONSOLIDATED EDISON COMPANY OF NEW YORK (Indian Point Unit 2) - POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point Unit 3)

Date of Proceeding: April 5, 1983

Docket Number: 50-247 SP and 50-286 SP

Place of Proceeding: White Plains, New York

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Raymond DeSimone
Official Reporter

Ruth Bennett
Official Reporter

RYTA RONCHER
Official Reporter