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50-247
50-286

October 6, 1982

Mr. Steven A. Varga
Chief, Operating Reactors Branch No. 1
Division of Licensing
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Varga:

In response to your September 15, 1982 letter, enclosed are our initial comments on the preliminary draft Letter Report on Review and Evaluation of the Indian Point Probabilistic Safety Study, Sandia National Laboratories, August 25, 1982 ("Sandia Report"). The following issues are addressed:

Attachment 1:	Wind;
Attachment 2:	Seismic;
Attachment 3:	Fire;
Attachment 4:	Pressurized Thermal Shock;
Attachment 5:	Fan Coolers;
Attachment 6:	Completeness (Sandia Report at 4.7-1); and
Attachment 7:	Important Findings (Sandia Report at 5.1-1 through 5.1-7).

As contemplated in your letter, we are prepared to continue our dialogue on the Sandia Report.

Sincerely,

John D. O'Toole
Vice President
Consolidated Edison Company
of New York, Inc.

J. Phillip Bayne
Senior Vice President
Power Authority of the
State of New York

A001

RESPONSE TO SANDIA LETTER REPORT OF SEPTEMBER 1, 1982 ON THE INDIAN POINT PROBABILISTIC SAFETY STUDY

October 1, 1982

Contents

ATTACHMENT 1 - WIND

The major concerns expressed by the Sandia comments relate to: (1) basic data used to develop the wind hazard curves; (2) the lack of consideration of site roughness characteristics and wind channelization effects along the Hudson River Valley; (3) the influence of adjacent buildings on the wind pressures on critical buildings; and (4) the fragility assigned to the offsite power system. These points are each addressed below.

1. WIND DATA

The primary issue appears to be the difference in the Batts, Russel, and Simiu 1980 published hurricane risk (Reference 30, IPPSS Section 7.9.5) and the site specific data presented in IPPSS Section 7.9.5. Batts' results were used by the reviewers to calculate the wind hazard. While the Batts data, for example, corresponds to an annual exceedance probability of 7×10^{-3} for 100 mph hurricane winds, the Indian Point study predicts an exceedance probability of 2.5×10^{-4} for 100 mph hurricane winds. Thus, at 100 mph this corresponds to a relative frequency difference factor of about 28. At this point, it seems that a systematic explanation of this difference is required to substantiate the Indian Point hazard curves. We have initiated the following additional activities:

1. Document differences in the Ho (Reference 43, IPPSS Section 7.9.5) data and the current HURDAT record maintained by the National Hurricane Center on a storm by storm basis. Identify multiple counting, smoothing techniques, and other assumptions made by Ho vis-a-vis the 1886-1980 HURDAT record.

2. Include wind direction in the simulation model. This will eliminate some of the conservatisms in the fragility analysis based on peak winds from all possible directions. It will also allow for variations in the surface roughness in the plant vicinity.

3. Assess sensitivity of the hazard curve to the hurricane windfield model and quantify these uncertainties.

4. Use the HURRISK code with the Ho data and Batts' windfield and check agreement with Batts' results. If there is close agreement, then the differences are in the data and not the models.

5. Perform sensitivity analysis and include formal uncertainty analysis on input pdfs to provide additional verification data for the hurricane risk curves.

2. SITE ROUGHNESS CHARACTERISTICS AND WIND CHANNELIZATION EFFECTS

It was pointed out by the reviewers that there is a potential for increased wind velocities at the site due to channelization effects along the Hudson River Valley. The reviewers indicated this might

result in a 10 mph or so increase, which corresponds to an increase in the hazard frequency by a factor of 5 or 10, depending on where you are on the curve. This question should be considered in conjunction with the methods used in the PRA that account for terrain roughness in the reduction of basic windspeeds to wind pressures. The approach would be to review topography, establish directional flow regimes and terrain roughness, and use continuity relations to predict channeled wind speeds on the river. By building this into the HURRISK computer model, the joint windspeed-wind direction frequency would include the river channelization effects. The importance of this effect will be offset somewhat by the inclusion of wind direction and the fact that channelization will not add uniformly to the risk at all windspeeds. At higher winds, it may be more important on segment 1 and less important for segment 2 (compare Figure IV-5, IPPSS Section 7.9.5), which dominates. This analysis has been initiated in order to refine the wind hazard model and curves.

We feel that it is highly unlikely that the wind risk estimates would increase by the significant factor being suggested by the reviewers, particularly when we look at all the elements in the analysis, quantify wind direction, further establish the data base, and assess model and windfield sensitivity in the uncertainty analysis.

3. STRUCTURES, WIND LOADING AND FRAGILITY

The reviewers expressed concern about the wind fragility of adjacent buildings (Unit 1 superheater building, Unit 2 turbine building) and the possible effect of their failure on Unit 2 critical structures. The influence of nearby structures is an important consideration in determining wind loads on structures. In the IPPSS, this was accomplished by selecting what were considered to be upper and lower bounds in wind loading from these effects. As stated in the analysis, the variability in effective velocity pressures that might be applied to the structures was determined through use of the ANSI A58.1 tables of effective velocity pressures. These tables provide values that are based on building exposures varying from open country, flat coastal belts, and grasslands to centers of large cities and very rough, hilly terrain. The effective velocity pressures between these two extremes at various building heights of interest varied by as much as a factor of 3. These extremes were incorporated in examining the possible wind loading on the buildings, including such variations over the range in building heights.

The potential wind failure of the Unit 1 superheater building and Unit 2 turbine building was considered early in the IPPSS analysis. These buildings were judged to have greater wind resistance than the steel buildings modeled because they have masonary walls which (experience indicates) are less susceptible to wind failures than steel framed buildings with metal siding and roofing.

4. OFFSITE POWER SYSTEM

In the IPPSS, the wind fragility of offsite power was established from onsite and near Buchanan substation transmission line towers and line

design criteria. The reviewers suggest that the transmission lines supplying offsite power from well beyond the Buchanan station might have less capacity. Without further investigation, we concur but do not believe it will significantly affect results.

5. FUTURE ANALYSIS

The reviewers' indicated factor of 20 greater core melt frequency from hurricane generated events at Unit 2 follows mostly from their judgment of use of nonconservative hazard curves, a greater accuracy required in the structures' wind fragility analyses considering effects of adjacent buildings, and a more conservative offsite power supply fragility. We believe our results will not be significantly affected by additional analysis. However, we have implemented an additional wind events analysis as follows:

1. Wind hazard curves analysis as discussed earlier.

2. New wind fragility analysis of:

   - Unit 1 - Superheater Building
   - Unit 2 - Turbine Building
   - Unit 2 - Control Building
   - Unit 2 - Diesel Generator Building
   - Unit 2 - Steel Portion of PAB
   - Unit 2 - Steel Portion of Auxiliary Feed Pump Building

   This analysis will consider shape factors, adjacent structures' influences on pressure loading on a directional basis, roofing and siding failure susceptibility, and siding and roofing fastner failure.

3. Modification of the events models to include the effects of failure of the turbine and superheater buildings on adjacent structures and facilities, and the concession of a loss of offsite power.

4. Recalculate core melt and release category frequencies based on any modified wind hazard curves, Booleans, and structure fragility curves that result from the new analyses.

ATTACHMENT 2 - SEISMIC

Principal seismic analysis concerns expressed by the reviewers were: (1) omission in the Indian Point Probabilistic Safety Study (IPPSS) of greater consideration to the Ramapo Fault as a seismic source zone; (2) the nonconservatism of the Woodward-Clyde Consultants seismic hazard curves (IPPSS Section 7.9.2); (3) the understatement of uncertainty in the fragility analysis; (4) the omission of the Unit 3 control room ceiling from the analysis; and (5) the lack of a detailed fragility analysis for the Unit 3 diesel fuel oil tanks. These are each discussed below.

The reviewers concluded that the impact of their concerns was that core melt frequency may be low by a factor of 2 for Indian Point 2 and by a factor of 8 for Indian Point 3. In light of the following analysis, we do not agree with the seismic core melt frequency evaluated by the reviewers.

1. RAMAPO FAULT

Two questions have been raised regarding the Ramapo Fault and its influence on the IPPSS study results:

- What is the seismic hazard at the site if the Ramapo Fault zone(s) is assumed to be a source area?
- What is the effect upon the integrated results if the Ramapo Fault zonation(s) is included separately or integrated in combination with other source zones that you examined in your integrated results?

In response to the above questions, a small source area centered about the Ramapo Fault zone was examined to determine the contribution of such an area to the seismic hazard at the Indian Point site (pages 6 and 8, Section 7.9.2, External Events). Because of the low number of seismic events reported for the Ramapo Fault zone, it is not possible to construct a meaningful frequency curve specific to the fault zone. The 51 events reported within the Ramapo source zone consist primarily of recent earthquakes of small magnitude which were instrumentally detected; only 9 events from this area were reported in the historical record prior to installation of a dense network of seismic stations in 1975. Of the 42 earthquakes recorded since 1975, only 10 exceed the magnitude threshold (M_b 2) for uniform coverage throughout the Ramapo source area. These data range in intensity between MMIII and MMV; the range in magnitude is restricted to between M_b 2 and M_b 2.8. No cumulative frequency curve that would be valid for the larger earthquakes of consequence to this study can be constructed from these data; both the number of events and their range in size are insufficient for this purpose.

To describe seismicity within this zone about the fault, therefore, the regional recurrence relation must be applied. The following answers to the questions reflect this result:

1. The seismic hazard at the site, if the Ramapo Fault zone(s) is assumed to be a source area, is the same as that presented in the study.

2. Because the cumulative frequency curves describing seismicity for a regional area must be assumed for the proposed Ramapo Fault zone, there is no change in the calculated result as presented.

Finally, in response to the statement preceding the two questions, the range of possible source areas needed for an adequate probabilistic study has been completely investigated. The IPPSS (Section 7.9.2) presents seismic data for the preferred source (source area 1) and tests the influence of other source area dimensions. Variations in source dimension were found to be insignificant in terms of calculated probabilities of ground motion. As shown in the initial study (Section 7.9.2, Figures 5 and 7), the preferred source area provides a slightly higher level of seismicity than the other areas tested. Based on the data presented, there is no justification for using multiple source areas to calculate probabilities for various ground motions at the site.

In addition to the relative absence of seismic events along the Ramapo Fault zone, other important kinds of information also indicate that the Ramapo seismic zone cannot be distinguished as a distinct seismic source area. A reexamination of regional indicators of tectonic stress, especially focal mechanism solutions (Reference 1), does not support northwest/southwest thrusting on the Ramapo Fault despite previous reports to the contrary (Reference 2). In addition, a recently completed geological examination of the Ramapo Fault zone (Reference 3) found no evidence that the fault is associated with current seismicity. Reference 3 states "Although drilling has confirmed the southeast dip of the Ramapo Fault, no evidence of reverse faulting was observed." The following discussion includes, therefore, results of recent studies of the tectonic stress regime near the Ramapo Fault, geological studies of the fault itself, and the seismological basis for the above responses.

Consideration of Seismic Areas

The first question asks that the Ramapo Fault zone be examined as a separate seismic source area. The difficulty in responding exactly as requested derives from the low level of seismic activity reported for the Ramapo Fault zone. That is, treatment of the Ramapo Fault zone as a separate seismic source area requires that specific seismicity parameters be derived based on reported epicentral data near the fault zone. Such data, however, are sparse and too limited both in number and in magnitude to derive seismicity parameters. Thus, neither historical nor instrumental observations provide a basis on which the Ramapo Fault zone can be delineated as a separate source area. There is no alternative but

to use seismicity parameters derived for a larger regional source area, and this procedure was adopted in the earlier study (source area 1, Section 7.9.2). Data that justify these conclusions are discussed below; a prior study (Reference 4) which these conclusions contradict is also addressed.

Selection of source areas for seismic evaluation in the IPPSS was guided by two criteria: (1) seismic activity throughout the area should appear uniform; and (2) the contemporary tectonic environment and geologic structure should be reasonably similar throughout the area. The goal of these criteria is to define an area within which there is a uniform likelihood of earthquake occurrence. Application of these selection criteria is not unequivocal and experts differ as to the best choice of source areas. Thus it was considered important to test the sensitivity of the results to various choices that have been proposed.

Specific source areas that are typical of the many tectonic provinces proposed for the region about the Indian Point site were selected to test their influence on calculated probabilities of ground motion. This comparison demonstrated an insignificant change in the calculated probabilities for the various choices (Section 7.9.2, Figures 5 and 7). The preferred source area (source area 1) which meets the selection criteria for a valid source area is characterized by the highest rate of occurrence of earthquakes per unit area of the choices evaluated (see Figure 5, Section 7.9.2). Therefore, the source area selected in the PRA report is either as conservative or more conservative than other proposed source areas in regard to rate of earthquake occurrence. The seismicity within this region consists of a coherent group of earthquakes that is sufficient in number, size, and range to construct a meaningful cumulative frequency curve.

A seismic source area centered on the Ramapo Fault was constructed and its ability to meet the two criteria for source area selection evaluated. The source area bounds the aggregate group of faults loosely termed the Ramapo Fault zone. It has a dimension in plan of approximately 125 kilometers by 20 kilometers. Earthquake data were again compiled for this zone using the same historical record as for the preferred source area in the PRA study. In addition, earthquakes that have occurred since the PRA study was completed were included in this data set.

Fifty-one earthquakes occurring between 1885 and 1982 have been documented within the identified zone about the Ramapo Fault. Forty-two of these earthquakes occurred since 1975; their detection is largely the result of the dense configuration of seismic stations installed beginning in 1975 in response to questions regarding the seismic safety of the Indian Point power generating facilities. Of the 42 events, 10 are reported to have magnitudes of about M_b 2 or greater. A magnitude level of about M_b 2 is the threshold above which earthquakes are thought to be uniformly detected in the New York City region (Reference 5). The detection of earthquakes of lesser magnitude is dependent upon their proximity to microseismic monitoring stations scattered throughout the area.

During the 200 to 250-year period of record prior to the installation of the dense array of seismic stations beginning in 1975, nine earthquakes have been reported within the proposed source area. Five of these earthquakes were felt; of these, four were determined to be of intensity MMIII, and one was determined to be about intensity MMIV. For the other four earthquakes, no intensity is reported. This suggests that these four earthquakes, all occurring since 1962, were instrumentally detected by nearby seismic stations operated by the Lamont-Doherty Geological Observatory of Columbia University and were not widely felt by the general population within the epicentral areas. Since 1975, intensities have been reported for six earthquakes. Of these, one had an intensity of MMIII, two had intensities of MMIII to MMIV, one had an intensity of MMIV to MMV and two had intensities of MMV. These relatively large intensities for earthquakes with magnitudes less than M_b 3 may be the result of the thorough intensity surveys currently conducted for earthquakes within the region. On a regional basis, earthquakes for which both intensity and magnitude values are reported suggest that earthquakes of about M_b 3 correlate with intensity MMIV (see Figure 6, Section 7.9.2).

A cumulative frequency curve for the Ramapo source area cannot be determined based on the available data. There are only 10 events with magnitude greater than M_b 2, the magnitude level necessary for uniform earthquake detection within the source area (Reference 5); all these magnitudes lie within a narrow range, 2.0 to 2.8. A cumulative frequency curve determined from such a narrowly based data set would not be meaningful, especially if extrapolated to magnitudes producing significant ground motion effects at the Indian Point site. Hence, to examine the seismic hazard at the site resulting from a Ramapo source area, it is necessary to use a regional cumulative frequency curve as a basis for predicting seismic activity.

If the cumulative frequency curve used to describe the seismicity within the preferred source area were to be applied to the Ramapo source area, we would expect to see an intensity MMIV earthquake within the zone on an average of about one every 40 years, an intensity MMV about every 114 years, and an intensity MMVI about every 300 years. Hence, we should expect up to five or six intensity MMIV events within the past 230-year period (but with a good chance that many went unreported or undetected), two to three intensity MMV events, and perhaps no events of intensity MMVI. Values of expected earthquake frequency are strictly based on the regional recurrence relation (normalized per unit area) as applied to the Ramapo source area.

The data suggest that seismicity within the Ramapo Fault zone source area about matches the expected seismicity level predicted by the regional curve. It follows, however, that the proposed Ramapo Fault zone cannot be distinguished from the larger source area. Hence, there is no basis in the seismicity of the area for consideration of a Ramapo source area as a separate source area.

As stated in the study (Section 7.9.2), the historical seismicity near the Ramapo Fault is far below that predicted by Aggarwal and Sykes (Reference 4). For example, the recurrence relation proposed there for the Ramapo Fault zone predicts the occurrence of an intensity MMV or greater earthquake about every 8 years for a total of 25 such events in the past 200 years. Only two such events are reported, however, within the Ramapo source zone during the complete historical period of record. This represents an order of magnitude fewer events than the curve proposed by Aggarwal and Sykes predicts. In fact, only 21 such events have been reported within the entire regional source area (source area 1, Section 7.9.2), which is about eight times as large as the area defined as the Ramapo Fault zone source area. A similar disparity exists for events of intensity MMVI and greater. The curve proposed by Aggarwal and Sykes (Reference 4) predicts events of MMVI and greater occur on the average of once every 25 years. No such events have been reported with the Ramapo source area. Based on the historical seismicity within the region, there are no data to support the recurrence relation suggested by Aggarwal and Sykes and no data justifying the identification of a separate zone of coherent seismicity within the Ramapo source area.

Tectonic Environment and Regional Stress

The present regional stress regime as inferred from geologic data, direct stress measurements, and fault plane solutions does not favor reactivation of northeast trending structures. Most of this data indicate that the maximum compressive stress is consistently oriented in a northeast direction. Data that support this conclusion are discussed here; recent investigations that do not agree with this conclusion (References 2 and 4) are also addressed.

There are considerable geologic data for the northeastern United States, and New York State in particular, that indicate large horizontal compressive stresses exist within the earth's crust. As reported by Sbar and Sykes (Reference 6), rock squeeze and horizontal shortening of bridge abutments and canal walls have occurred near Rochester, Lockport, and Niagara Falls; in some instances the shortening has been of substantial magnitude. Similar phenomena have been reported in adjacent parts of Canada for north trending tunnels; these have not been observed for east trending tunnels (References 6 and 7). Squeeze phenomena in deep openings have been observed in New York City (Reference 8) where accompanying stress measurements indicated a compressive stress ranging from 110 bars to 280 bars in magnitude and from N18^{o}E to N64^{o}E in direction. Deep stress measurements made using hydrofracture techniques in western New York and Pennsylvania (Reference 9) indicate similar northeast compressive stresses of about 220 bars with directions of N77^{o}E and N70^{o}E, respectively.

Further support for this regional stress environment is suggested by some fault plane solutions. In northern New York and adjacent parts of Canada, the 1966 Attica, New York earthquake (Reference 10), the 1975 Maniwaki, Ontario earthquake (Reference 11), the 1973 Altona,

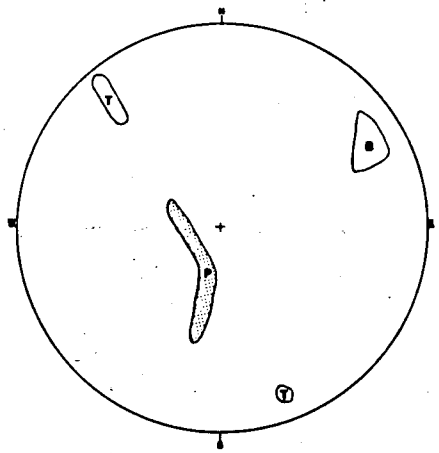
New York earthquake, and the 1971 to 1973 Blue Mountain Lakes earthquakes (Reference 12) all suggest a northeast direction of maximum compressional stress (N62°E, N50°E, N73°E and N71°E, respectively).

In southeastern New York and northern New Jersey, the 1974 Wappingers Falls earthquakes (Reference 13) and the 1969 Lake Hopatcong earthquakes (Reference 6), both within the preferred source area for this study, have reported compressional (P-axis) stress directions of N51°E and N56°E, respectively (Reference 12). More recently, similar determinations have been reported for several earthquakes (1977 through 1980) that occurred near Annsville, New York (References 14 and 15) just northeast of the Indian Point site.

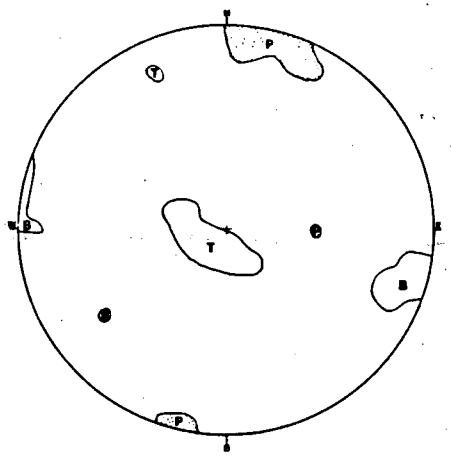
References 2 and 4 present fault plane solutions for the region about the Ramapo showing thrust faulting along planes trending northeast. These solutions are used to propose a southeast/northwest direction of maximum compressional stress for the region, as well as to postulate the reactivation of northeast trending structures in response to the applied regional stress. The proposed reactivation of northeast trending structures is directed specifically toward the Ramapo Fault as well as other major northeast trending structures.

Based on reexamination of earthquake data for which first motions are available from published network bulletins, we are unable to reproduce the results presented in References 2 and 4. Work in progress (Reference 1) shows compressional (P-axis) stress directions inconsistent with many of the fault plane solutions presented by Yang and Aggarwal (Reference 2). Hence, reactivation of the Ramapo Fault is not supported by available earthquake data as presented, and regional stress directions as inferred from earthquake data do not support the proposed high angle reverse or thrust faulting along generally northeast trending planes.

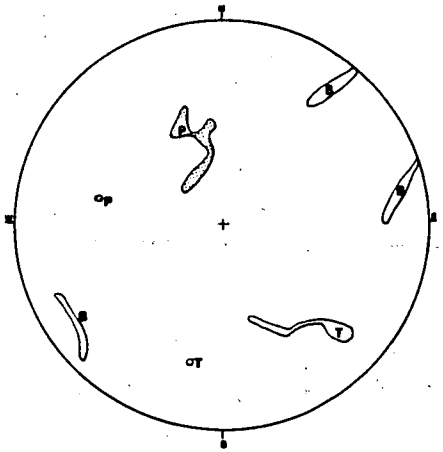
A larger data set of earthquakes for the New York City area has been examined and indicates that a compressional (P-axis) stress direction is consistently northeast trending for the majority of the well recorded earthquakes. This investigation maps the orientation of acceptable P-axes using existing network data from published bulletins. These data are processed using a modified version of the focal mechanism program of Guinn and Long (Reference 16). This method compiles in a consistent manner all valid fault-plane solutions. With this method, however, it is difficult to incorporate significant variations in data quality. Nodal arrivals or questionable first motions must either be included with equal weight to other readings, or excluded. Data quality and character can, however, be used to choose preferred solutions from those that are consistent with the data. Figure 1 shows the domain of P on the lower focal hemisphere using the least number of inconsistent data points possible to obtain a valid solution(s). First motions are not used for stations with arrival time residuals greater than three-quarters of a second. All other available station data were used in the determination of the maximum compressive stress domains presented.



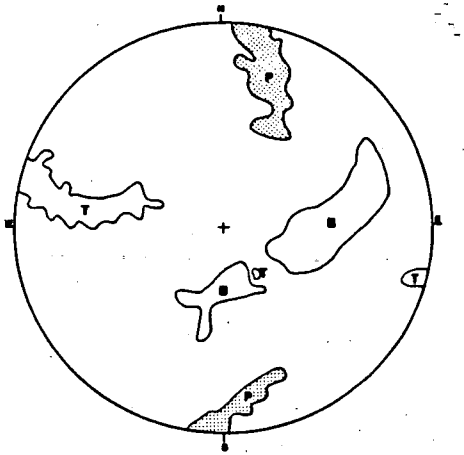
POMPTON LAKES EVENT
11 MARCH 1976
DEPTH – 6.22 KM



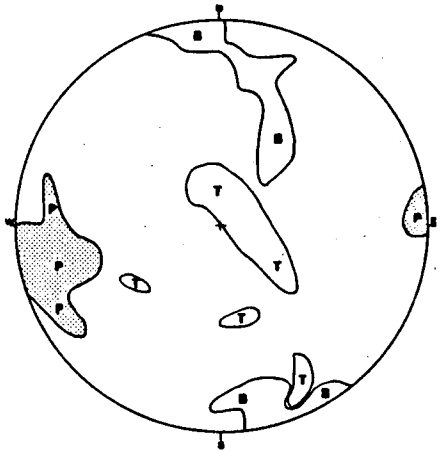
TUXEDO PARK EVENT
10 MARCH 1977
DEPTH – 8.74 KM



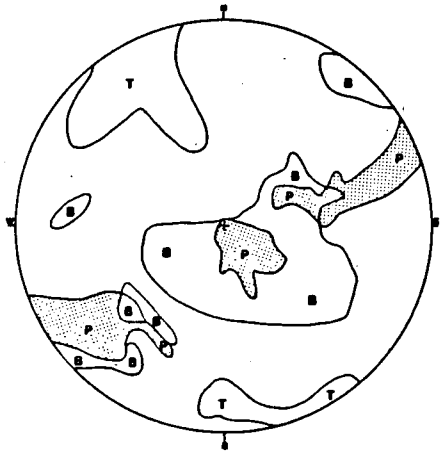
RIDGEFIELD EVENT
18 APRIL 1976
DEPTH – 3.4 KM



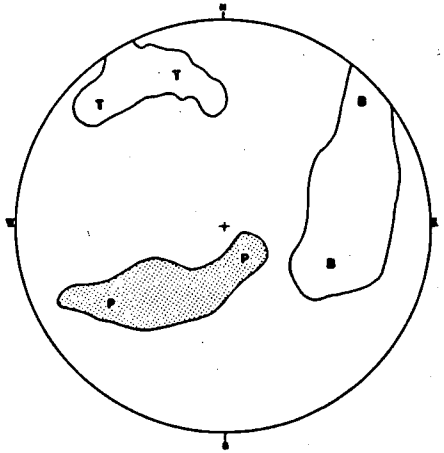
PEEKSKILL EVENT
23 JULY 1978
DEPTH –



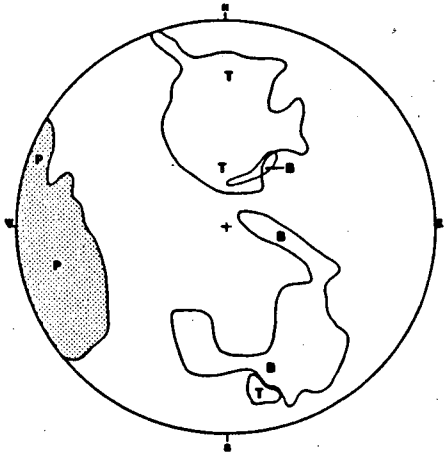
BENARDSVILLE EVENT
10 MARCH 1979
DEPTH – 5.61 KM



MT. PLEASANT
20 AUG 1976
DEPTH –



ANNSVILLE EVENT
25 JAN 1978
DEPTH – 1.54 KM



ROCKLAND LAKE EVENT
10 MARCH 1982
DEPTH – 12.15 KM

FIGURE 1 (page 1 of 2)

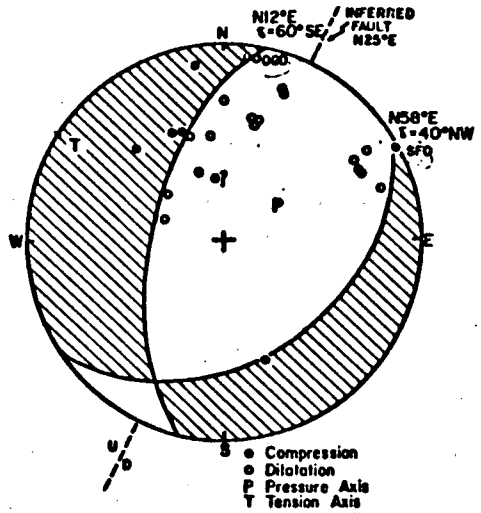
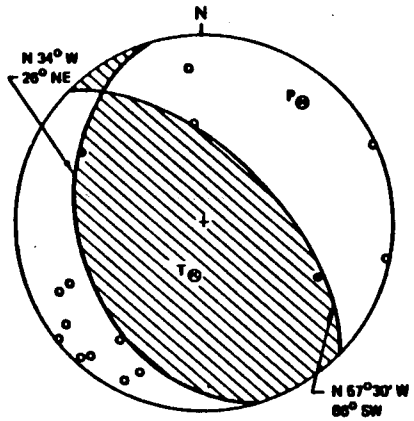


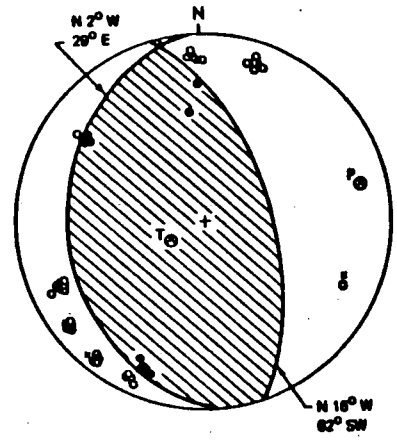
FIG. 8. Composite fault-plane solution from 13 microearthquakes and four felt sh four felt shocks are combined into one point each for OGD and SFO. The diagram is an projection of the upper hemisphere of the radiation field.

LAKE HOPATCONG



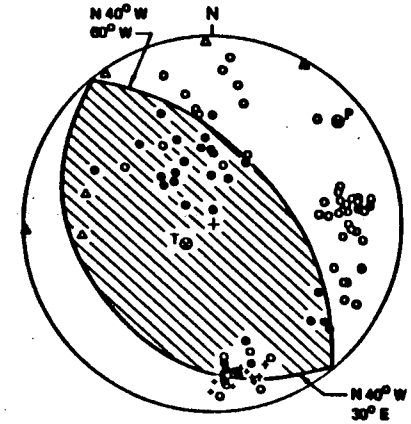
a

ANNSVILLE
1977



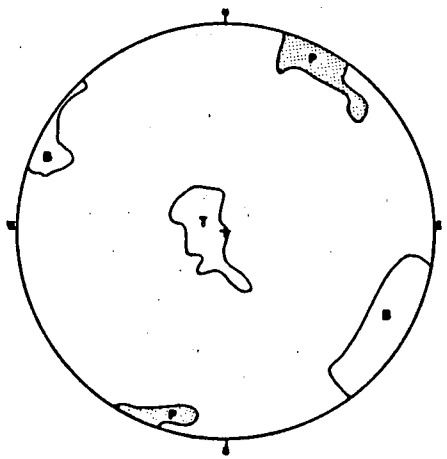
b

ANNSVILLE
1980

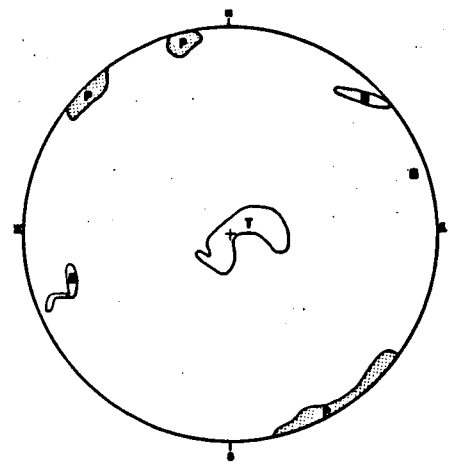


c

WAPPINGERS FALLS
1974



THORNWOOD EVENT
4 SEPT 1980
DEPTH – 8.9 KM



INDIAN POINT EVENT
22 SEPT 1976
DEPTH –

FIGURE 1 (page 2 of 2)

Based on results presented in Figure 1, there appears to be good agreement between the majority of available local earthquake data and the direction of regional maximum compressive stress inferred from other data. Indeed, the composite data set suggests a reasonably uniform style of fault movement for microearthquakes within the region. This movement is consistent with northeast compressive stress.

The presence of northeast compressive stress is inconsistent with thrust type movements on the northeast trending Ramapo Fault zone or other subparallel structure. On the contrary, expected motions produced by northeast compressive stress are similar to the motions inferred for earthquakes such as the Wappingers Falls sequence, several earthquakes near Annsville, the Rockland Lake earthquake, the Suffern-Tuxedo Park earthquake, and the Thornwood earthquake. Further work may reveal other examples of this type.

This conclusion does not suggest that the Ramapo Fault, or other major northeast trending geologic features, has no role in the spatial distribution of seismicity. These major geologic structures while trending parallel to the regional compressional stress may serve as zones of structural weakness along which stress is concentrated. Thus, at least some earthquake occurrence along planes normal to major northeast features may be the result of stress concentrations by these features.

In summary, a reevaluation of available stress data indicates that the regional stress regime throughout the northeastern United States, specifically including the New York City region, is inconsistent with reactivation of northeasterly trending structures. Upon reexamination, the data upon which Yang and Aggarwal (Reference 2) have based their conclusions regarding northwest trending maximum compressional stress do not support their conclusion. Thrust type motions on planes trending normal to the Ramapo Fault are favored by the regional compressional stress that consistently trends northeast. This conclusion provides further support for not considering the Ramapo Fault zone as a separate source area.

Geologic Characterization of the Ramapo Fault

There are no geological data resulting from recent detailed examination of the Ramapo Fault zone (Reference 3) that suggest the fault is associated with current seismicity. As stated by Ratcliffe (Reference 3),

> Although drilling has confirmed the southeast dip of the Ramapo Fault, no evidence of reverse faulting was observed. The mineral textures and structures seen in the cores are compatible with Sinemurian or post-Sinemurian (lower Jurassic) normal faulting. No data require faulting more recently than Sinemurian, although such an interpretation cannot be ruled out.

The data that support these conclusions are discussed here.

Field investigations indicate that the Ramapo Fault is part of a group of northeast trending, southeast dipping faults. The group of faults lies in a zone approximately 5 km wide, known as the Ramapo Fault zone. The faults range in age from Precambrian through Mesozoic (Reference 16) and cut Precambrian crystalline rocks and Mesozoic igneous and sedimentary rocks. The geological significance of the Ramapo Fault lies in the fact that it represents the boundary of the Triassic-Jurassic Newark Basin and brings Mesozoic sedimentary and igneous rocks into fault contact with Precambrian crystalline basement. Other Mesozoic faults within the fault zone may be as significant as the Ramapo Fault with respect to total displacement. However, total displacement along these faults is unknown since they occur within the Precambrian rocks and do not offset recognizable stratigraphic units.

Many of the faults, including the Ramapo, have had long and complex geologic histories (Reference 17). These faults originated in Grenville time and experienced intermittent reactivation through at least early Jurassic time (References 3 and 16). Another group of faults developed during Paleozoic orogenesis, and another during the Mesozoic. Not all pre-Mesozoic faults were reactivated during the Mesozoic.

Faults of different ages can be distinguished by the texture developed along slip surfaces. Pre-Mesozoic faults are ductile to semiductile, and are characterized by low to mid-greenschist facies blastomylomites and mylonites (References 3, 16, and 18). In addition, the older faults have steep dips and are deep (15 km) basement features. Mesozoic faults are characterized by brittle deformation that has produced unconsolidated clay gouge and cataclasite, and mineralization typical of low temperature conditions (References 3 and 16). The younger faults have moderate to steep dips near the surface, but are thought to flatten out with depth (Reference 3).

The Ramapo Fault was intercepted by drill cores in two locations near Sky Meadow Road in Suffern, New York, where rocks of the Newark basin are in fault contact with basement rocks (Reference 3). The actual fault surface, recovered from the core, dips 55° SE, and is represented by a 5-cm thick layer of clay gouge derived from Triassic basalt and Precambrian gneiss. The gouge is an indication of younger (Mesozoic) faulting, and in places crosscuts older mylonite features in the gneiss. Textural evidence from the gouge and from adjacent layers indicates repeated faulting, with normal faulting being the last movement on the fault (Reference 3). Microscopic textural evidence, and the gross distribution of rocks in the area indicate that the Mesozoic movement on the northeast trending southeast dipping faults was right-oblique normal. No geologic evidence has been recognized to date that warrants reverse faulting.

Calcite-chlorite-laumontite mineralization is associated with the clay gouge and is found as vein fillings in both the hanging wall and footwall blocks. Ratcliffe (Reference 3) attempted to determine the age of laumontite from veins in the hanging wall with the K-Ar method. An age of 32 ± 3.8 m.y. was determined, but the age is not considered reliable

since $^{40}Ar/^{39}Ar$ analyses performed on the laumontite in the gouge suggest that some of the system was not closed with respect to radiogenic Ar. In addition, some of the laumontite in the gouge is deformed, indicating that some movement on the fault postdates mineralization. No conclusions can be made, based on the age, regarding maximum or minimum ages for movement along the fault (Reference 3).

Microscopic and megascopic geological observations indicate that the last motion along the Ramapo Fault was normal faulting. If the Ramapo is being reactivated, the most probable plane of failure would be the unconsolidated clay gouge. Thus, if reverse faulting was presently taking place along the fault, it would occur along a surface predisposed to preserving textural evidence of such movement. No such evidence has been recovered from the fault to date.

Summary

In summary, the seismicity data (both historical and recent instrumental records) have been reviewed in detail. The existing seismic data do not permit the identification of a small source area centered on the Ramapo Fault zone. The reported seismic events within such a zone are too few in number and too small in magnitude and range to allow derivation of a recurrence relationship. Based on current geologic studies, there are no data to suggest that the Ramapo Fault zone has shown recent movement. There are no data that suggest the Ramapo Fault zone is currently being used to accommodate movements in reponse to regional tectonic stresses. Reexamination of indicators of current tectonic stress consistently suggests northeast trending axes of compression. This regional stress orientation is not consistent with thrust faulting on reactivated northeast-trending structures such as the Ramapo Fault.

2. SEISMIC HAZARD CURVES

The reviewers indicated that the Woodward-Clyde Consultants (WCC) seismicity study (IPPSS Section 7.9.2) gives results which are too unconservative, therefore, they chose to predict core melt frequencies using just the Dames & Moore study results (IPPSS Section 7.9.1). As in all areas of the risk assessment, we have used all the information available to express our state of knowledge. Even though the two consultants' results differ, these results represent expert opinion and we consider it inappropriate to ignore the work of one in preference to the other. In particular, three main comments were made by the reviewers regarding the Woodward-Clyde Consultants study. These comments directly question the completeness of the study results to the IPPSS. The comments were:

- All reasonable hypotheses considered explicitly in the analysis were not included in a statement of the uncertainty in the frequency of exceedance curves.

- Important alternative hypotheses were not considered in the analysis; specifically, consideration was not given to a Ramapo Fault zone.

- The uncertainty and the mean of the distribution on the upper bound intensity should be increased.

Consideration of Alternative Hypotheses

An investigation of the sensitivity of the results to alternative hypotheses for input parameters forms part of the study. It was found that the effect on the frequency of exceedance curve of alternative hypotheses can be classified according to three categories.

In the first category, the effects on the frequency of exceedance curves of considering alternative hypotheses were small (less than a factor of 2) in the range of interest. In this case, a weight of 1.0 was assigned to the most conservative hypothesis. In the second category, visible alternatives produced significant effects (more than a factor of 2) on frequency of exceedance curves within the range of interest. In this case, alternatives are weighted commensurate with scientific judgments. In the third category, alternative hypotheses can cause significant effects, but only one hypothesis is considered applicable to the site region. In this case, the single, reasonable hypothesis applicable to the site region was given a weight of 1.0.

In conclusion, the study represents a probabilistic assessment of the seismicity of the site region. When appropriate, alternatives for various parameters were weighted commensurate with scientific judgments. In cases for which reasonable alternative hypotheses could not be identified, or in which consideration of multiple alternatives produced an insignificant effect on the frequency of exceedance curves, single parameters were assigned a weight of 1.0.

Consideration of the Ramapo Fault Zone

A seismic source zone centered about the Ramapo Fault was specifically identified in the seismic exposure analysis conducted for the IPPSS. Only a small number of historical earthquakes of maximum intensity MMIV are reported within this zone. The set of recent instrumental data consists of 10 earthquakes above the uniform detection threshold of about magnitude 2, with a maximum magnitude of about 2.9. Based on these data, no meaningful seismicity parameters can be identified that distinguish this zone from the surrounding area in source area 1. Seismicity parameters for the Ramapo Fault zone must therefore be assumed to be the same as those used to describe the seismicity of the larger region (source area 1). For this case, contributions to seismic risk at the site resulting from the Ramapo Fault zone are identical to contributions from the subregion of source area 1 that coincides with the Ramapo Fault zone. Hence, the effect of the Ramapo Fault zone is implicitly included as a subarea of source area 1. Detailed evaluation of the Ramapo Fault zone and its potential contribution to regional seismicity were presented earlier.

Consideration of Upper Bound Intensity

An upper bound earthquake was included as a required parameter to describe regional seismicity. A composite value for this parameter was selected after review of pertinent geologic, tectonic, and seismologic data. The chosen value has been reevaluated in light of concern expressed in the draft review of the IPPSS; namely, that the frequency of exceedance curves do not adequately represent the seismicity and seismic risk of the site region.

WCC found no basis or scientific evidence to suggest that an upper bound earthquake of greater size than that originally identified should be used to characterize the seismicity of the Indian Point site region. The proposed magnitudes for historical earthquakes inferred from intensity data (in the Sandia draft letter report) are adequately accounted for in the estimates of the upper bound earthquakes for the region. Review of geologic and tectonic data for the region, including recent data obtained and compiled regarding regional tectonic stress and geologic characteristics of specific structures such as the Ramapo Fault, support the selection of the identified upper bound earthquakes. The composite upper bound earthquake selected (80% likelihood of intensity VII, 20% likelihood of intensity VIII) adequately defines WCC's scientific judgment and conclusions, and appropriately bounds its uncertainty with respect to characterization of regional seismicity and seismic risk. In WCC's judgment, no additional conservatism is warranted in characterizing regional seismicity.

3. FRAGILITY UNCERTAINTIES

It was indicated by the reviewers that the uncertainty of the fragility parameters used in the seismic analysis are understated, but the median values are conservative. The fragilities used in the analysis include a measure of the random variability, based on the potential range and value of the many parameters individually considered. In addition, uncertainties, β_U, relating to the earthquake characteristics were assigned. Together, both constitute the large overall uncertainty applied to the fragilities.

Certainly, in many cases, subjective but expert opinion must be applied in making best estimates of the margin of safety when considering each of the capacity and response characteristics of structures and equipment in order to obtain the best estimate, or median, acceleration capacity of these items. Since they were derived from best estimates of contributing parameters, they are considered to be reasonably valid. Similarly, the random and uncertainty variability for each of these parameters was used to determine the random and uncertainty variability about the median acceleration capacity.

Randomness includes variabilities which cannot be substantially reduced based on the current state of the art of seismic analysis and material behavior knowledge. Included in the values assigned to randomness are variabilites in material strength characteristics, earthquake

characteristics, and estimates of modal analysis SRSS with absolute sum responses. These variabilites were based for the most part on plant specific data or data from other representative nuclear power plants. Uncertainty includes estimated variabilites from lack of knowledge of a given parameter. Uncertainty could normally be expected to be reduced, in some cases substantially, by further analysis or test. Typical sources of uncertainty include variabilities from modeling assumptions and use of nonsite specific or generic data.

As an example, the factor of safety for the strength contribution to the overall fragility evaluation of a containment wall shear failure (IPPSS Section 7.9.3, Table 4-1) considered both the steel and concrete material properties as well as modeling uncertainties. Plant specific results of concrete cylinder tests or reinforcing steel tests were not available for the IPPSS. Based on data for other nuclear power plants, a lognormal standard deviation (β) for concrete f'_c, including aging, was determined to be approximately 0.13. Similarly, β for reinforcing steel strength was estimated to be 0.09. Using the second moment approximation from statistics, these values were combined to give a β for material properties alone of 0.05. However, the random variability in the ultimate strength of a concrete shear wall results from other sources in addition to just the strengths of the steel and concrete. Among these sources are the number of cracks and crack patterns, variations in bond splitting and local crushing around individual rebars, in-place versus cylinder concrete strengths, strain rate effects, and cyclic response characteristics. A lognormal standard deviation of 0.09 was estimated for these effects which was combined with the 0.05 material strength value to give a β for randomness for the strength factor of 0.10.

Variabilities included in the uncertainty lognormal standard deviation include contributions from the modeling errors expected in the use of the shear wall strength equation, load distributions in the shell at response levels near failure, effective concrete area, interaction between the liner and reinforcing steel, flexure in the shell, and use of nonsite specific material properties for the strength determinations. The lognormal standard deviation associated with uncertainty was estimated to be 0.20.

4. CEILING FRAGILITY

It was indicated by the reviewers that the seismic analysis did not consider the failure of the ceiling in the control room when assigning a fragility to the control building. The seismic analysis did consider the control room ceiling. It was judged that falling panels might result in injury to some personnel in the control room, but certainly not incapacitate all the operators or result in a loss of equipment functions in the room. In view of the review comment, we have investigated the Unit 3 ceiling fragility and the effects of its possible failure (see Appendix 1). Using the median seismic hazard curve and median fragility curve for the control room ceiling, the analysis concludes that, considering the requirement to incapacitate all operators in the control

room, the seismic contribution to core melt is about 3.6×10^{-6} per year for Unit 3. This value is similar to the value obtained in the IPPSS of 3.1×10^{-6}. The ceiling consideration increases the core melt contribution from seismic events by a factor of 2.3. This is not significant compared to the core melt frequencies obtained for all initiators.

5. UNIT 3 DIESEL FUEL OIL TANK FRAGILITY

The reviewers suggested that the Unit 3 diesel fuel oil tanks dominate the seismic core melt frequency and, as such, warrant a more detailed fragility analysis. These components do not dominate a release category or core melt as seen by the Booleans in IPPSS Section 7.2.5 and the acceleration capacity of several of the components within the Booleans. In cases such as this where a number of components are the major contributors without a single component dominating, a nominal change in the capacity of one component has little impact on the results. The median capacity assigned to the diesel fuel oil tanks was believed to be conservative so additional calculations are likely to principally reduce the uncertainty. To pursue these points, computations were previously accomplished to evaluate the worth of additional structural analysis. It was assumed that further analysis of the tanks could show that the seismic capacity of the tanks could be sufficiently high that they would not have to be considered failing from any of the postulated earthquakes. A calculation of the change in core melt frequency was then performed with the tanks removed from the Booleans. The result was an annual core melt frequency of 2.3×10^{-6} as compared with the earlier results of 3.9×10^{-6}, or about a one-third reduction. On that basis, no further analysis is justified.

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APPENDIX 1

LOGIC FOR INCAPACITATING INDIAN POINT 3 CONTROL ROOM PERSONNEL VIA SEISMIC INITIATED CEILING FAILURE

Reference: Structural Mechanics Associates Analysis, attached.

Postulates:

1. Some transite ceiling panels may dislodge since they are not fastened to their supports, resulting in their falling onto the aluminum eggcrate ceiling panels below and then both falling into the control room.

2. Transite panels which do not fall with a leading edge or corner at the operator's location will bruise, but not incapacitate, an operator. Falling eggcrate panels alone are not likely to incapacitate operators.

3. Three operators are located in the control room and all three must be incapacitated in order to lose control.

4. Loss of control leads to core melt.

Data:

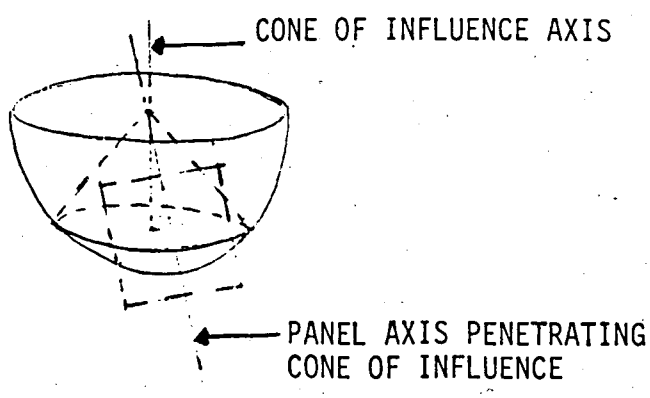
Transite panel weight: 1/4-inch x 3-feet x 4-feet	25 pounds
Eggcrate panel weight: 3-feet x 4-feet aluminum eggcrate	8 pounds
Total weight:	33 pounds

Given failure of ceiling panels (as per seismic hazard and fragility information):

1. Probability of top panel failure causing falling of eggcrate and top panels = 0.8.

2. Probability of transite or eggcrate panel falling with a damaging edge intersecting a horizontal plane:

   a. Surface of hemisphere = $2\pi r^2$.
   b. Surface of segment = $2\pi rh$.
   c. For right cone, $h = r - 0.707 = 0.293r$.



d. Surface of right cone segment = $2\pi r(0.293r) = 0.586\pi r^2$

e. Fraction of hemisphere surface = $\frac{0.586\pi r^2}{2\pi r^2} = 0.293$, say 0.30

	F_i	p_i	$p_i F_i$
Distribution on bounds:	0.25	0.2	0.05
	0.30	0.5	0.15
	0.50	0.3	0.15
			0.35

F = 0.35 = fraction of panel axes falling within area of influence.

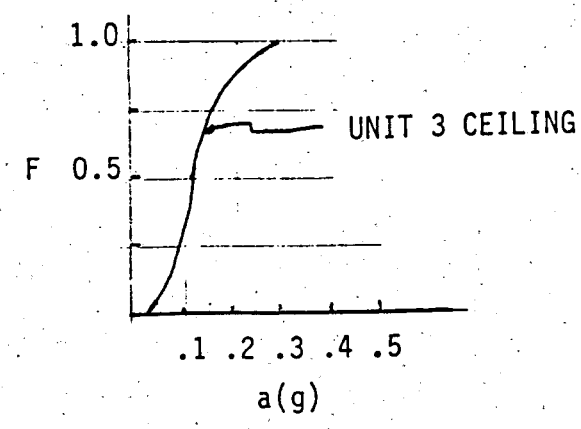
3. Conditional p of panel over operator being a failed panel = 0.6 (considering dependencies).

4. Conditional p of one operator being incapacitated = 0.8 x 0.6 x 0.35 = 0.168.

5. Conditional p of three operators being incapacitated = $0.168^3 = 0.0047$.

Calculation of core melt frequency (using composite fragility curve and median seismic hazard curve):

Ceiling Capacities: Unit 3 $\tilde{\tilde{a}} = 0.11g$ $\beta_C = 0.52$



Discrete Values Acceleration (g)	Discrete Values Ceiling Capacity	x 0.0047	Fraction of Earthquakes Causing Loss of Control
0.05	ε	0.0047	ε
0.10	0.30	0.0047	1.4×10^{-3}
0.15	0.70	0.0047	3.3×10^{-3}
0.20	0.87	0.0047	4.1×10^{-3}
0.25	0.95	0.0047	4.5×10^{-3}
0.30	1.00	0.0047	4.7×10^{-3}
0.35	1.00	0.0047	4.7×10^{-3}
0.40	1.00	0.0047	4.7×10^{-3}

Seismic Hazard (median)		Unit 3	
Acceleration	Frequency	F	Annual Core Melt Frequency
0.05	-	ε	ε
0.10	1×10^{-3}	1.4×10^{-3}	14.0×10^{-7}
0.15	3×10^{-4}	3.3×10^{-3}	9.9×10^{-7}
0.20	1.5×10^{-4}	4.1×10^{-3}	6.2×10^{-7}
0.25	8.0×10^{-5}	4.5×10^{-3}	3.6×10^{-7}
0.30	5.0×10^{-5}	4.7×10^{-3}	2.3×10^{-7}
0.35	ε	4.7×10^{-3}	ε
0.40	ε	4.7×10^{-3}	ε
Total			3.6×10^{-6}

SEISMIC FRAGILITY ANALYSIS OF THE INDIAN POINT UNIT 3 CONTROL ROOM CEILING

The Unit 3 control room ceiling system utilizes light fixture hangers typically consisting of Unistrut channels bolted to continuous Unistrut concrete inserts embedded in the slab above. One-quarter inch thick transite panels are supported by flanges of the light fixtures. The eggcrate panel ceiling below is supported by an aluminum tee-bar grid which, in turn, is hung from the light fixtures by 1/4-inch diameter rods. Perforated aluminum acoustical panels typically span between the tee-bar grid to the structure perimeter.

The light fixtures which support the 1/4-inch thick transite panels are bolted to the hangers. The tee-bar grid supporting the eggcrate louvers is hung from the light fixtures by rods. Other light fixtures located flush with the eggcrate ceiling are hung from the Unistrut concrete inserts by Unistrut channel sections.

The transite panels bear on the light fixture flanges without any positive, mechanical connections. If sufficient horizontal accelerations due to seismic response of the ceiling system occur, the frictional resistance between the panels and the light fixture flanges will be overcome and the panels will slide. Because the transite panels were to have been cut and installed in the field, it is reasonable to believe that the gap between the panels edges and the vertical faces of the light fixtures may exceed the overlap between the panel and the light fixture flange. If sliding occurs, the panel may be expected to drop.

This mode of failure for the Unit 3 control room ceiling has the lowest acceleration capacity of the failure modes evaluated. It has a median acceleration capacity of approximately 0.11g with logarithmic standard deviations associated with randomness and uncertainty of approximately 0.30 and 0.42, respectively.

The occurrence of this ceiling failure mode is dependent on the worst case assumption that there is sufficient gap between the panel edge and the vertical light fixture face to overcome the panel support overlap during sliding. If sufficient gap is not available, the control room ceiling would be expected to have a higher acceleration capacity.

ATTACHMENT 3 - FIRE

The reviewers stated that cable damage prior to ignition due to exposure to a layer of hot gases was not considered in the analysis. We cannot confirm or reject the reviewers' revised estimate for the frequencies of accident scenarios at present; further analysis of the cable tray tests by Underwriters Laboratories (UL) is required in order to determine the effects of the hot gas layer damage mode.

We note that the change in accident frequencies may not be significant because the UL test results to date are not directly applicable to Indian Point Units 2 and 3.

1. The UL tests were conducted in a relatively small room. The temperature of a hot gas layer in the Indian Point compartments analyzed would be lower due to the larger size of the Indian Point room.

2. The damaged cable trays in the UL tests were much closer to the test chamber ceiling than the critical Indian Point cable trays are in their respective rooms. The hot gas layer temperature decreases with distance from the ceiling.

3. The UL source fire was larger than that used in the IPPSS, and its frequency of occurrence would be somewhat lower.

4. The accuracy of the final results can only be assessed if the impact of conservative as well as nonconservative assumptions is quantified. The treatment of various heat sinks is an example of an area where the analysis has been conservative.

More refined experiments will help reduce the uncertainties in fire analysis. Furthermore, the reviewers have noted that the Appendix R modifications have not yet been modeled in the IPPSS studies. The modification, which may include alternate cables placed in locations not affected by the fire, would minimize the effect of hot gas layers.

ATTACHMENT 4 - PRESSURIZED THERMAL SHOCK

With respect to the overall pressurized thermal shock (PTS) issue, Reference 1 addressed the major concerns in this area from a probabilistic viewpoint.

WCAP-10019 (Reference 1) and a subsequent report (Reference 2) transmitted to the Commission by the Westinghouse Owners Group provide accident analysis results and a generic assessment (including probabilistic risk assessment) of the reactor vessel integrity problems for all Westinghouse plants. The analyses have included both operational transients and design basis events which result in a cooldown exceeding the allowable rate limits, and/or which result in inside vessel wall fluid temperatures below 300°F, including potential for system repressurization. These analyses provide a conservative assessment of the years of operation prior to exceeding vessel integrity acceptance criteria. Further evaluations and discussions have been held between the Westinghouse Owners Group (WOG) and NRC, and an NRC position and required actions will be established. Once this is done, more specific answers to the accompanying PTS questions could be provided. However, the numerical results submitted in Reference 2, along with discussions held between Westinghouse and the NRC, indicate that the values assumed for vessel failure beyond ECCS capability in the IPPSS are in close agreement. Thus, the contribution to overall risk from PTS in this study is not expected to change based on recent NRC discussions.

During certain refueling outages which are stated in the technical specifications, vessel material specimen capsules which have been subjected to the same neutron flux as the vessel wall are removed from the reactor vessel. These specimens are then tested to determine the effects from radiation upon their physical properties; i.e., fracture toughness. From this analysis, heatup and cooldown curves are generated to lessen the effects of cyclic thermal stresses, thus reducing the probability and/or severity of a pressurized thermal shock event.

Additionally, the present fuel cycle incorporates a modified low leakage loading pattern which will result in a reduction of a fast neutron flux in the periphery of the core, with the subsequent effect of decreasing embrittlement of the reactor vessel walls.

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ATTACHMENT 5 - FAN COOLERS

FAN COOLER AND SPRAY SURVIVABILITY

The effects on the containment sprays and fan coolers of conditions in the containment during a degraded core accident have been assessed. The conclusion is that the fans and sprays would likely survive these conditions. This is due to the conservative design criteria utilized for these systems which result in a significant design capacity. This design capacity coupled with the expected conditions in containment would be of sufficient margin to allow for confidence in continued successful operation of the systems. Following is a discussion of our assessment of the containments safeguards ability to operate for various degraded core conditions. For both systems (i.e., the fan coolers and sprays), both the effects of hydrogen burn and aerosol loading have been considered. The fan coolers are discussed first and then the spray system.

During an energetic hydrogen burn, the containment pressure can be expected to rise by about 60 psi in about 20 to 60 seconds and then decay to the preburn value in a short time. This is a slower pressure rise than that for the design basis accident, which results in a pressure rise of about 47 psig in about 10 to 20 seconds, and for which the fan coolers and containment spray systems in Indian Point Units 2 and 3 are designed. Because of the inherent conservatism in the design analysis, it is expected that the fan coolers and spray systems should be capable of withstanding a significantly more severe pressure transient than this design basis transient. An example of this type of conservatism is that the containment vessel can withstand about 2.5 times its original design value.

In the case of the spray system, it is designed to withstand LOCA pressures and basically is a system of piping and pumps that one would expect to have a very high tolerance for any expected pressure surges in the containment. Hence, the fan coolers and spray systems are expected to survive the pressure transient during a realistic hydrogen burn scenario.

The duration of thermal transient from a significant hydrogen burn would be short (1 to 2 minutes). Components of fan cooler and spray systems are relatively large, rugged in construction, and hence should have substantial thermal inertia. During equipment survivability tests conducted at Fenwal, Inc. and Acurex Corp., the peak surface temperatures measured for equipment with relatively low thermal inertia (such as the pressure transmitter and limit switch) exposed to hydrogen burns were generally under 300°F. As the components of the fan cooler and spray systems have substantially greater thermal inertia than the small equipment tested in the above tests, their peak temperatures during hydrogen burns can be expected to be below 300°F. All components of the fan cooler and spray systems are designed to withstand, without impairing operability, a post-accident containment

temperature of 271°F for 3 hours. Because of conservatism in the design analysis, the fan cooler and spray systems can be expected to operate at temperatures substantially above 271°F for a few minutes. The peak component temperature anticipated during hydrogen burns (about 300°F) is only slightly higher than the design basis (271°F) and its duration (a few minutes) is substantially shorter than that assumed for the design basis transient (3 hours). Hence a short duration thermal transient from a hydrogen burn should not have adverse effects on heat removal capability or fan operation.

Several experimental and analytical studies have been made by the industry to assess the effects of hydrogen burns on electrical equipment. These studies have shown that electrical equipment can perform its intended function during and after being exposed to hydrogen burns. Electrical equipment in Indian Point Units 2 and 3 is expected to be similar in design to equipment already tested or being tested by the industry. Based on the data available to date, electrical equipment at Indian Point Units 2 and 3 is expected to survive the environmental conditions created by hydrogen burns.

During a degraded core accident, aerosols are generated and released to the containment atmosphere. This generation could pose a threat to the operation of the fan cooler systems.

We have performed a limited assessment of the effects of aerosols on filter efficiencies, crud buildup on cooling coils, and fan cooler efficiency. The effect is expected to be minimal, considering that:

1. The HEPA filter efficiency is at least 99.97% during accident operation and the accumulation of aerosols on the filters could only increase their removal efficiency by self filtering. The filters are not expected to fail by aerosol loadings that could exist inside the fan coolers under degraded core conditions.

2. Crud would not build up on cooling coils because of condensed water that constantly washes out the aerosols trapped by the cooling coils. The heat removal capacity of the cooling coils will not be affected by release of aerosols into the containment. It is worth noting that both Indian Point plants have just completed replacement of cooling coils. The Unit 2 RCFC will have 6% greater capacity than the original equipment and Unit 3 will have 20% greater capacity.

3. The aerosol loadings on the HEPA filters were estimated to cause only 2.2 inches w.g. pressure drop across the filters when five fan coolers were assumed to be operating in Indian Point Unit 2. This pressure drop would not have a significant effect on the fan cooler efficiency. Aerosols would not affect Indian Point 3 fan cooler performance because of a 26,000 cfm bypass flow.

Aerosols could be generated during core degradation in-vessel or from core/water/concrete interaction in the reactor cavity. Gap and melt releases are largely responsible for release of fission products and core materials from the primary system while the vaporization release during core melt/water interaction and subsequent concrete attack (if it occurs) is responsible for ex-vessel aerosol generation and release into the containment.

The aerosols generated in-vessel, which consist of fission products and other core materials, would be effectively removed in the pathways; i.e., core and internal structures, piping, to the containment by natural deposition (either laminar or turbulent), and gravitational settling. Since the steam content in the reactor vessel is high, fission product removal by steam condensation is also significant. The retention of particulates in the primary system depends on the thermohydraulic conditions involving a core melt. For accident sequences with low steam flow rates in the primary system, such as transient or small break events, a small fraction of the total aerosol mass from the core would be released from the primary system. Given the long and winding paths and low steaming rates for most of the accident sequences following core melt, it is expected that a very small fraction of the aerosol inventory in-vessel will escape into the containment.

Of the aerosols that get into the containment, aerosol plateout, agglomeration, and gravitational settling will quickly remove the aerosols from the atmosphere and prevent them from reaching fan coolers and containment spray headers near the dome. In addition, if the containment sprays are operating, considerable airborne aerosols would be washed out of the atmosphere. Within the RCFCs, the moisture separators would remove significant quantities of particulates prior to reaching the HEPA filters.

The quantity of aerosols generated during core degradation is primarily from core melt. As stated in the IPPSS, in the case where water is available to form a coolable debris bed following vessel failure, very little aerosol would be released from the core debris since the core melt is quickly quenched. Even if one assumes that the debris bed is noncoolable, the basic plant geometry ensures that the core debris will be water covered when fan coolers are operating. This water cover will remove most of the aerosols prior to their release into the containment. Therefore, whether a coolable or noncoolable debris bed is formed, aerosols generated from core/concrete interaction are not an issue.

The fan cooler units are located on the intermediate floor (Elevation 68') between the containment wall and the primary compartment shield walls. The fan coolers take suction from the immediate surroundings and discharge to the distribution duct work. To reach the fan cooler suction, airborne aerosols must take a tortuous path from the primary system, up through the operating deck grating, to the lower containment outside the shield wall. It is expected that a large fraction of the aerosols will settle out in the transport path and will not be able to reach the fan cooler suction.

The above arguments that support our view of an insignificant effect of aerosols on the fan coolers are summarized as follows:

1. Aerosol generation would be limited to those generated in-vessel.

2. Aerosols would be mostly retained in the transport pathways, such as primary systems piping and structures, water in the reactor cavity and the containment sump, etc.

3. The fan cooler location, associated ducting, and moisture separators will preclude aerosols reaching the HEPA filters in a significant amount.

Even if we ignore the above arguments, it can also be shown that the HEPA filters can absorb signficant quantities of aerosols without significantly affecting the fan cooler performance. In Section 3.2.2.6 of the NUREG-0850 report, a maximum of 3,080 pounds of areosols was extrapolated from highest available experimental results. It should be noted that this estimate is much higher than that given in the NUREG-0772 report. As we discussed above, a very small fraction of aerosols would reach the HEPA filters. For this analysis, we conservatively assume that 25% of the 3,080 pounds of aerosols would reach the HEPA filters. This amount of aerosols would cause only 3.1 inches and 2.2 inches w.g. pressure drops if three and five fan coolers were assumed to be operating, respectively, at Indian Point Unit 2.

Aerosol plugging would not be a problem for the Indian Point 3 fan coolers since each fan cooler has a 26,000 cfm flow which bypasses the HEPA filters during an accident. Even if the HEPA filters were plugged, the fan coolers would still maintain a flow rate higher than 26,000 cfm in each unit, which is adequate to guarantee successful operation.

As far as the effect of aerosols on spray operation is concerned, our conclusion is that there would not be significant adverse effects. This conclusion was based on the fact that aerosols by definition would not be present inside the spray train unless the system is void of water and therefore not in operation anyway. Therefore, the only concern would be any impact aerosols might have on the exposed exterior surfaces of the spray system. The impact would be minimal, considering that:

1. When the sprays are in operation, the sprays will wash off exterior surfaces of the system and reduce the aerosol concentration significantly. The spray nozzles would not be affected by airborne aerosols because they would be washed out before they reach the nozzles.

2. The spray system is designed for the harsh LOCA environment and the mechanical portions of the system consist primarily of steel piping and components which would not be adversely impacted by the aerosols.

3. The electrical systems, because of their LOCA design should be able to withstand the aerosol generated radioactive environment.

ATTACHMENT 6 - COMPLETENESS

Note: Sandia's comments are listed sequentially in this style type and single spaced. Responses follow in this style type at space and a half spacing as shown here. A numbering system has been added for aid in identifying comments.

4.7 COMPLETENESS

Section 4.7 of the Sandia Report indicated that:

> One of the major sources of uncertainty in any PRA is completeness. These types of uncertainties arise from the inability of the PRA analysts to completely identify all possible accident sequences and system failure modes. Our review identified several accident sequences and system failure modes which were apparently omitted in the IPPSS. The more important omissions are summarized below.

4.7.1 Pressurized thermal shock--discussed in Section 2.1 and not evaluated in this review.

For response, refer to Attachment 4 - Pressurized Thermal Shock.

4.7.2 Steam generator tube rupture coincident with a stuck open secondary safety valve--discussed and evaluated in Section 4.1.

A revised steam generator tube rupture analysis which will include the case of a stuck-open secondary safety valve is currently in preparation.

4.7.3 Hot gas layer failure mode safety system cabling during a fire--discussed and evaluated in Section 2.7.4.

For response, refer to Attachment 3 - Fire.

4.7.4 The Ramapo Fault was not considered as a source zone in the seismic analysis--discussed and evaluated in Section 2.7.1.

For response, refer to Attachment 2 - Seismic.

4.7.5 Safety System failure caused by core meltdown phenomena--discussed and evaluated in Section 4.2.

For response, refer to Attachment 5 - Fan Coolers.

4.7.6 An initiating event caused by a pipe break in the component cooling water system--discussed and evaluated in Section 4.6.

The initiating event caused by a major pipe break in the component cooling system (large enough to rapidly drain the CCS) was calculated by the reviewer to make no change in health effect risk, but a measurable increase in core melt frequency. This contributor to core melt frequency is believed to be a substantial overestimate for the following reasons: passive rupture in such a mode may be impossible for this low temperature, low pressure piping system; the system is well protected externally; and the time to substantial reactor coolant pump seal LOCA is conservatively estimated so additional time should be available for the operators to align alternate cooling water sources for critical pumps.

4.7.7 Wind channelization of hurricane winds--discussed and evaluated in Section 2.7.2.

For response, refer to Attachment 1 - Wind.

4.7.8 Low pressure system and containment spray system β factors were omitted--discussed and evaluated in Section 2.4.

For response, refer to Attachment 7 (5.1.6.4).

4.7.9 Reactor coolant pump seal ruptures were not included in the small LOCA initiating event data base--discussed and evaluated in Section 2.1.

Reactor coolant pump seal ruptures large enough to exceed the capacity of a single charging pump were conceptually included in the small LOCA initiating event data base. However, none occurred at other plants during the time period covered by the IPPSS generic data base or at Indian Point 2 or 3 up to the time of the study. Had later generic failures been included, the effect on the Indian Point Units 2 and 3 small LOCA frequency estimates would have been minimal.

4.7.10 Steam generator overfill scenarios were not considered--not discussed or evaluated in this report.

Steam generator overfill scenarios were considered during the IPPSS but were not discussed in the study. Three factors are involved in the judgment that such scenarios are not significant contributors to risk or core melt frequency:

- Overfill to the point of flooding the mainstream lines is very unlikely. For nonfaulted steam generator conditions, main feedwater isolation automatically occurs on high steam generator level. Thus, only auxiliary feedwater is available for overfilling. This assures that significant time is available for the operator to recognize the potential for overfill and control/isolate auxiliary feedwater. For faulted steam generator conditions, the same isolation occurs and the operator is explicitly instructed to isolate faulted steam generators. However, recognizing that under faulted steam generator conditions there is a higher potential for overfill, these scenarios will be explicitly treated in the revision to the steam generator tube rupture analysis.

- Failure of the main steam lines, even if flooded, is unlikely.

- Flooding or failure of a main steam line has very little effect on the progression of accident sequences; however, the frequency of certain scenarios can be increased. These types of scenarios include steam generator safety valve failure when passing liquid.

4.7.11 Unit 3 control room ceiling failure due to a seismic event--discussed and evaluated in Section 3.3.8.

For response, refer to Attachment 2 - Seismic.

4.7.12 Ground roughness and shape of building effects on wind dispersion--discussed in Section 2.7.2.

For response, refer to Attachment 1 - Wind.

ATTACHMENT 7 - IMPORTANT FINDINGS

Note: Sandia's comments are listed sequentially in this style type and single spaced. Responses follow in this style type at space and a half spacing as shown here. A numbering system has been added for aid in identifying comments.

5.1 IMPORTANT FINDINGS

Section 5.1 of the Sandia Report indicated that:

> among the important findings of our [Sandia's] review are the following, grouped by topic:

5.1.1 INITIATING EVENTS

5.1.1.1 [Page 5.1-2] The initiating events covered in the IPPSS seem to be relatively complete compared to those addressed in previous PRAs, and their estimates of initiating event frequencies appear reasonable.

No response required.

5.1.1.2 [Page 5.1-2] An exception to this was found. The initiating event of a pipe break in the component cooling water system was not considered. This was analyzed by us in Section 4.6.

Refer to Attachment 6 (4.7.6).

5.1.1.3 [Page 5.1-2] The initiating event frequencies for each plant are based on the operating history of each plant.

No response required.

5.1.2 EVENT TREES

5.1.2.1 [Page 5.1-2] The treatment of the containment spray system (CSS) is questionable. The IPPSS assumes that the CSS can be used throughout an accident in the injection mode rather than having to draw from the sump. They assume that the operator will act to conserve the water in the refueling water storage tank by sparingly operating the pumps and that, if depleted, the tank can

be refilled. If it is not refilled, and LPRS subsequently fails, sprays would not be available to mitigate the accident consequences.

Because electric power is available in such situations, there is a high probability that the fan coolers will be operable. Therefore, containment integrity would be maintained without sprays. In addition, there are alternative ways of providing water to the containment sprays. The plant analysis section of the IPPSS contains event trees which include a nodal question regarding the functioning of the containment spray system in the recirculation mode. Such functioning involves the operation of the recirculation pumps, drawing water from the containment recirculation sump and pumping through the residual heat removal heat exchangers to the spray ring headers. Close examination of the plant state assignments for the branch points in the event trees will reveal that, even for cases where such operation succeeded, no credit in terms of plant state assignment was taken for recirculation sprays. Instead, only where injection phase spray succeeded was the "C" designator in the plant state applied.

This approach is essentially a simplification relative to plant event state development and represents a degree of conservatism in the study. For example, an SEF plant state means that the containment spray system failed to function during the injection phase. It is applied to both the success and failure branches of the recirculation spray question. Therefore, even if the recirculation spray system succeeds, no credit is taken for spray operation.

The application of the "C" designator to all sequences where injection phase spray operation succeeded is not unrealistic. Even without fan cooler operation, the Indian Point containment can tolerate significant pauses in spray operation without approaching overpressure failure limits. For example, given an SE plant state, the time to overpressure failure is at least 10 to 12 hours. The refueling water storage tank (RWST) has enough of a "heel" of water at switchover to recirculation to permit injection spray operation for about 30 minutes. This permits the

sprays to reduce any initial pressure rise. The Indian Point study examined (but did not set forth in the report) the ability of plant operators to refill, to some reasonable level, the RWST in the 10 to 12 hour time period prior to containment overpressure. Several sources of water exist which could be used for this purpose.

Operation of the spray system to prevent overpressure containment failure does not need to be a continuous function as noted earlier. Approximately 10 to 12 hours are required to fail the containment in the absence of all containment heat removal. If only one of the methods of filling the RWST is employed, enough water can be added to the RWST in about 3 hours to allow the spray system to effectively hold the pressure well below the failure level. Such a process can continue on an intermittent basis until either the fan coolers or the RHR recirculation system can be placed into effective operation. Considering the time available, the frequency of the failure of such actions is very low.

Because containment failure will not occur for at least 10 hours, there will be a full complement of emergency support personnel to implement recovery actions. Therefore, although there are no post-accident RWST refill procedures at Indian Point, it seems likely that this will be accomplished. Additionally, given the diversity of means available to supply water to the spray system, it seems reasonable to assign a low likelihood of failure to the long term operation of the spray system.

5.1.2.2 [Page 5.1-2] Operator recovery actions (such as the one noted above) were often assumed to be performed with negligible failure probability. This assumption appears to be overly optimistic.

See the preceding response.

5.1.2.3 [Page 5.1-2] Core melts caused by overpressure failure of containment (e.g., S_2C type accidents in WASH-1400) were not considered. However, this would have negligible effect on risk.

This scenario was considered. However, it was not considered a core melt sequence because the Indian Point recirculation pumps are designed to operate under such conditions. In addition, the core cooling and one of the containment heat removal systems share most of the same equipment (i.e., pumps and support systems). Hence the probability of having core cooling and not containment heat removal during recirculation is negligible compared with other contributors to core melt.

5.1.2.4 [Page 5.1-2] Feed and bleed capability is given more credit than the procedures indicate it should.

Following the TMI accident, Westinghouse performed a variety of generic analyses to demonstrate the inherent safety of Westinghouse designed PWRs. Some of these analyses included loss of coolant accidents induced by the complete loss of all feedwater. The complete loss of all feedwater will result in a small LOCA through the pressurizer PORVs as the secondary decay heat removal degrades. These analyses were reported in WCAP-9600, "Report on Small Break Loss of Coolant Accidents in Westinghouse NSSS Systems" and WCAP-9744, "Loss of all Feedwater Induced Loss of Coolant Accident Analyses." The purpose of those analyses was to investigate the minimum operator action time in the event that no feedwater was available for the removal of decay heat. WCAP-9915, "PORV Sensitivity Study for LOFW-LOCA Analyses," reports the results of a number of sensitivity studies to this type of analysis and provides an indication of the relationship between PORV capacity, steam generator dryout time, and operator action time.

The loss of feedwater induced LOCA is a small LOCA through the pressurizer PORVs. The analysis method utilized the small break LOCA evaluation model, WFLASH, which is described in WCAP-8200 entitled "WFLASH: A FORTRAN IV Computer Program for Simulation for Transients in a Multi-Loop PWR." The conservative Appendix K inputs were used, such as 120% of ANS decay heat. WFLASH has been verified and approved by the U.S. Nuclear Regulatory Commission for use in conservative evaluation of small LOCAs in a Westinghouse PWR. Numerous sensitivity studies have been performed to verify the input of the WFLASH model for small LOCAs.

The loss of feedwater induced LOCA analysis was initialized so that the steam generator water level used in WFLASH resulted in a steam generator dryout time which matched calculation by an accepted independent process. The WFLASH loss of feedwater LOCA analysis uses conservatively low liquid and vapor heat transfer coefficients and secondary metal heat capacity is not modeled to reduce secondary heat removal capability. The pressurizer PORVs were modeled as pressure dependent break flow paths to the containment. The mass flow rate through the PORVs is based on the rated PORV flow which is used to determine an appropriate flow area for use with the evaluation model break flow rate. This methodology will result in a conservatively short operator action time.

Westinghouse has developed procedures for the recognition and consequential mitigation of a complete loss of feedwater. This procedure was submitted for NRC review in December 1981. It has been included in the emergency response guideline and is being integrated into the critical safety function monitoring. Westinghouse has also held training seminars on this issue.

The loss of fluid test facility near Idaho Falls, Idaho, performed experiment L9-1/L3-3 on April 15, 1981. This experiment was designed to examine the decay heat removal capability in the complete loss of feedwater scenario. The L9-1/L3-3 loss of fluid test terminated the main feedwater and waited until there was a high hot leg temperature before tripping the reactor. This provided a very conservative inventory on the secondary side, and was more limiting than the Indian Point criteria in that respect. Additional decay heat removal capability extended the time scale of events just as a longer dryout time would. Most of the transient behavior predicted in WCAP-9600 and 9744 was observed. The pressurizer PORV showed decay heat removal margin and successful operability under this scenario condition. Westinghouse believes a detailed modeling of the loss of feedwater LOCA scenario would also show additional operator action time margin for Westinghouse PWRs. The WCAP-9600 and 9744 calculations were designed to be conservative.

Westinghouse does not believe that this test should be used to verify the WFLASH model due to loss of fluid test atypicalities, but believes the WFLASH model has sufficient verification for conservative analyses of this nature.

Based on the Westinghouse analyses, sensitivity studies, and the additional decay heat removal capability demonstrated in the Loss of Fluid Test Facility, Westinghouse believes that primary bleed and feed is a viable mode of decay heat removal in the event of a loss of all feedwater in the Indian Point nuclear units. In fact, the reviewers concur with our assumption as stated on page 4.3-1 of the Sandia Letter Report:

> It should be noted that we feel that feed and bleed core cooling should be given credit. Recent tests at the LOFT facility and Westinghouse analysis suggest that feed and bleed is a viable core cooling option.

As pointed out by the reviewers, plant operators at both Indian Point Units 2 and 3 received feed and bleed simulator training.

5.1.2.5 [Page 5.1-2] As a result of our review, the steam generator tube rupture and ATWS event trees are being reconstructed. In IPPSS currently, the former contains errors, and the latter does not represent the as-built plant.

The steam generator tube rupture and ATWS event trees are being revised.

5.1.3 [Page 5.1-2] SUCCESS CRITERIA

Success criteria used in the analysis appear to be reasonable and consistent with those used in PRAs of similar plants.

No response required.

5.1.4 FAULT TREES

5.1.4.1 [Page 5.1-2] In general, the fault trees presented in the IPPSS are an accurate representation of the IP-2 and IP-3 systems. The analysis was considerably aided by the fact that the fluid

systems have common headers, thus making the construction of supercomponents much easier.

No response required.

5.1.4.2 [Page 5.1-3] The analyses are inconsistent in the application of common cause failure possibilities, not only among systems, or different modes of the same system, at the one plant but also for the same system in the other plant where no difference could be discerned. The IPPSS, however, should be commended for its examination of common cause failures although it treated them subjectively, rather than examining historic data.

Additional analyses of common cause failures are being conducted for a nuclear power plant similar to Indian Point, and will be examined upon completion. Risk is not expected to be affected.

5.1.4.3 [Page 5.1-3] In the degraded power states, the IPPSS often ignored maintenance unavailability for the pumps which could still receive power.

Pump maintenance was inadvertently omitted from the CCW analysis and the SW conventional header analysis for the single bus available case. However, this omission does not contribute significantly to risk. All other analysis included maintenance for all degraded power states.

5.1.4.4 [Page 5.1-3] In the sequence evaluations for the loss-of-offsite power initiating event, the wrong service water system unavailability was used.

The affected sequences have been requantified using the correct system unavailability. Negligible changes in IPPSS results were found.

5.1.4.5 [Page 5.1-3] The calculation of the low pressure recirculation system failure probability assumes that, should the recirculation pumps fail and the operator switch to the RHR pumps, the RHR system will work. The fault tree for the system, however, is right in that it considers RHR failure as well.

RHR system hardware unavailability was considered, however, it is not included in the IPPSS report because analysis confirmed its small

contribution in relation to the operator error rate in the switchover operation.

5.1.4.6 [Page 5.1-3] Several errors were identified in the analysis of the auxiliary feedwater system. However, their effect was shown to be of small importance.

The reviewers correctly pointed out an error in assuming turbine-driven AFW pump trip on loss of flow, even though this effect is of small importance. Regarding the operator action needed to bring up the turbine-driven pump speed, it is included as part of the operator actions required for pump startup (see cutsets 2, 3, and 4 of Table 1.5.2.3.9-8G). As such, this is implicitly modeled in our analysis.

5.1.4.7 The analysis of the interaction of the service water system with the containment fans and high pressure recirculation is wrong. As a result of this finding, the IPPSS analysts are revising the analysis.

This analysis has already been revised with negligible changes in IPPSS results.

5.1.5 HUMAN RELIABILITY ANALYSIS

5.1.5.1 [Page 5.1-3] The human reliability analysis reflected a diligent and sincere effort to use accepted human reliability analysis methods. Some general problems, however, were recognized. Among them were: undue optimism in the assessment of credit for human redundancy; optimistic assessments of human performance under stress, especially for the case of multiple problems;

In assessing human performance under stress, the IPPSS has taken into account the integrated effect of operator training, operators' familiarity with each particular sequence under study, anticipated operator responses, combined stress level due to all the events that have occurred, time dependent availability of the STA, time window for accident recognition, decision-making and operator actions, existing operating and emergency procedures, available instrumentation, and the competing demands for the operators' attention. As a result of the Director's Order, human factors evaluations of the control room layout

and emergency operating procedures at Indian Point were conducted. Additionally, since the operators are trained on a plant specific simulator, their familiarity with various accident scenarios should lower stress and improve response time in actual emergencies. Therefore, the estimate of operations performances in IPPSS is reasonable.

5.1.5.1.3 [Page 5.1-3] personal estimates of operator performance rather than using simple measurements;

Actual measurement of operator performance involved in responding to emergency plant operation is, in most cases, unrealistic and nonconservative. As a result, any in-plant simulation of this kind will usually underestimate the operator reaction/response time. This is mainly due to the lack of emergency stress level, simultaneous interfering events, and competing demands for operator actions in any attempt for this measurement. We believe that expert assessment of operator performance is the best way to include the preceding considerations and to obtain realistic estimates. As a specific example, consider the histograms of operator response over time in IPPSS Sections 1.3.2.2 and 1.3.2.3 which estimate the time required for restoration of offsite power supply. Much longer (and more realistic) times are used than one would observe by timing an operator walking through these actions.

5.1.5.1.4 [Page 5.1-3] inadequate documentation of the use of expert opinion;

The documentation of expert opinion was consistent with, and as comprehensive, as other PRAs performed by the consultants. Each scenario was studied in detail to determine the required operator actions, available instrumentation, conflicting concerns, and available response time. The study covered the context of personnel availability, procedural guidance, general training, physical locations of equipment, and historical experience relevant to the scenarios. Power Authority of the State of New York and Consolidated Edison operators, supervisors, engineers, and

analysts were consulted for their expert opinions relating to the scenarios. This information was then used to develop the histograms used in the study. Those are considered to be the best format for expressing the available state of knowledge regarding these scenarios. In addition, all the histograms thus developed were properly documented in the appropriate sections of IPPSS.

5.1.5.1.5 [Page 5.1-3] optimistic assessments of dependence among tasks done by the same person;

We believe that our assessment of dependence is neither optimistic nor pessimistic and appropriately reflects uncertainties. Dependent human errors were assessed for several scenarios in the systems analyses presented in Sections 1.5.2 and 1.6.2. The general dependence relations stated in Section 0.15.2 were used as a consistent basis for these assessments. The specific valve restoration example on page 1.5-123 of the IPPSS was not used in any of the systems analyses. Each analyst developed a dependent error model that was compatible with the specific testing, maintenance, or operator action scenarios for his system.

5.1.5.1.6 [Page 5.1-3] apparent nonconsideration of some possibilities for common cause failures from human errors;

Referring to page 2.5-8 of the Sandia Comments, the system in question is the reactor protection system (RPS). On page 2.4-3 of the Sandia Comments, the reviewer correctly describes the common cause quantification performed on the RPS. The two comments appear to contradict each other. To answer the question on page 2.5-8, common cause miscalibration of the RPS was quantified in the IPPSS in the paragraph quoted (page 1.5-389).

5.1.5.1.7 [Page 5.1-4] possible insufficient consideration of errors in restoring safety components after test, maintenance, or calibration.

The procedures used in IPPSS to evaluate potential human errors are:

1. Detailed review of all test and surveillance procedures associated with safety related equipment. This review enabled the system analysts to determine the following:

   a. The effect due to testing/surveillance for cases where system lineup is not changed.

   b. For cases where system lineup is changed, determine whether indications are available to detect incorrect lineups after test and whether incorrect lineups could be corrected without operator intervention due to automatic signals, etc. Frequencies of occurrence are then assigned to the possible human errors and to the recovery actions prior to system actuation.

   c. In addition, for those cases where human error could result in a misaligned train, the frequency of common cause errors (coupling of errors) which results in misaligned systems was calculated and its contribution was included in the frequency of system failure.

2. Review of post-maintenance testing and return to service procedures indicated the following:

   a. Post-maintenance checkout procedures require the operation of safety related equipment as the final check prior to returning the equipment to operational status.

   b. For equipment in normally operating systems, this check is a full flow test of the system.

   c. For standby systems, the test and surveillance procedures are used, and the effects of human errors associated with these procedures have already been identified.

In light of the preceding procedures, common cause failures from human errors and errors in restoring safety components after test, maintenance, or calibration were systematically identified and evaluated in IPPSS.

5.1.5.2 [Page 5.1-4] The failure to switchover to high pressure recirculation appears to have been overestimated in the IPPSS while the equivalent error for low pressure recirculation appears to have been underestimated.

The differences between the IPPSS and the Sandia estimates for failure to switchover to recirculation cooling is primarily due to the different modeling of operator recovery and of dependence level between the shift watch supervisor and the shift technical advisor. The reviewers took more credit for recovery through annunciators in the case of high pressure recirculation and took less credit for recovery from the two control board operators in the case of low pressure recirculation. In addition, the reviewers assumed a higher dependence level between the shift watch supervisor and the shift technical advisor in the case of low pressure recirculation. There is no effect on risk.

5.1.6 ESTIMATION METHODOLOGY

5.1.6.1 [Page 5.1-4] Indian Point's estimates of maintenance unavailabilities appear to be consistent with Indian Point data.

No response required.

5.1.6.2 [Page 5.1-4] The treatment of uncertainty associated with estimates from existing data sources is inconsistent. Generally, 5 and 95 percent bounds from WASH-1400 were used as 20% and 80% limits in IPPSS. Notable exceptions to this were the treatment of interfacing system LOCAs, pressure vessel rupture, and pipe ruptures. In all three cases, substantially higher estimates would have been obtained had their general rule been followed. The results are highly sensitive to this assumption. (It must be noted that the revised sequence analysis used the 5 and 95% bounds.)

The failure rate distributions used in the IPPSS reflect the study team's state of knowledge. Each case was evaluated individually and the reported distribution was the consensus of the analysts. In the process

of quantifying our state of knowledge, published evidence that experts tend to overestimate their knowledge (e.g., Lichtenstein, S., et al, "Calibration of Probabilities: The State of the Art," Reference 0-19 in the IPPSS) as well as our own experience with generic distributions (Nuclear Engineering and Design, Vol. 56, pp. 321-329, 1980) were taken into account. Therefore, the WASH-1400 team's curves, in most cases, were not accepted unconditionally but rather were used as "expert opinion" information input to developing our own prior probablility distributions. In order to reflect more uncertainty in our minds than in the WASH-1400 team's, we decided initially to broaden the distributions. This is a typical way to treat expert opinion. It should be pointed out that this broadening was only an initial step and that the resulting distributions still had to be compared with history and to withstand the scrutiny of the analyst. It turned out that the use of 20/80 percentiles resulted, in most (but not all) cases, in distributions that the team found acceptable. In addition, the results are not, in general, very sensitive to the choice of 20/80 percentiles except for a few cases. Given that most of these generic distributions are specialized using Indian Point specific data, the effect of the use of 20/80 percentiles is indeed insignificant, and did not in fact result in substantially higher estimates than would have been obtained had 5/95 percentiles been used.

In the particular case of interfacing system LOCAs, the initial quantification used the 20/80 distributions as presented in the methodology section. The results did not agree with the large amount of evidence which is available concerning this failure mode. They indicated a frequency of occurrence 3 to 4 orders of magnitude higher than the published results. Based on this initial quantification, a data search was begun to determine the frequency of failure due to disc rupture. During this search, several interesting facts were found.

1. There have been no failures due to valve disc rupture in the nuclear power industry.

2. Phone conversations with valve manufacturers revealed no gate valve disc failures to the best of their knowledge. Typically, valve manufacturers are required to hydrostatically test valve discs and bodies to 150% of design pressure. They knew of no failures during hydrostatic testing for any of their product lines.

Because of this data, it was felt that, in the absence of detailed experience (number of valves in similar locations in the industry and the exposure hours at pressure for each valve), the distribution from WASH-1400 adequately described our state of knowledge concerning this failure mode.

5.1.6.3 [Page 5.1-4] The Bayesian methodology used to estimate accident sequence rates is somewhat oversold, but it does have the positive effect of highlighting the importance of plant specific data. Where Indian Point data exist and are used to modify IPPSS's prior probability distributions, the effect of the prior distributions is generally unimportant with respect to estimated accident sequence rates. Where Indian Point data are not available or used, the estimates are quite sensitive to the assumed prior distribution.

The fact that the data tend to dominate the posterior (specialized) distributions is as it should be. When the Indian Point evidence is weak, the prior (subjectively assessed) distribution is important and the results are sensitive to it. The Bayesian methodology is a broad framework in which the risk analysis is performed. State-of-knowledge uncertainties are emphasized everywhere in the IPPSS, while "point" or "best" estimates are, in general, eliminated, as are other similarly ill defined concepts as "optimistic" and "pessimistic" analyses. The propagation of uncertainties is done through standard methods of combining distributions.

5.1.6.4 [Page 5.1-4] The inclusion of the O-factor for accounting for "other," dependent causes of failure is inconsistent.

One may disagree with specific numbers, but not with the fact that the category "other" does exist. At the time the IPPSS was performed, there was no generally accepted methodology for treating the "other" category.

5.1.7 EXTERNAL EVENTS

5.1.7.1 [Page 5.1-4] For the seismic hazard and fragility analysis, the methodologies used in the IPPSS are appropriate and adequate to perform a seismic risk analysis.

No response required.

5.1.7.2 [Page 5.1-4] The Ramapo fault zone was not included in the analysis and should be addressed.

See Attachment 2 for response.

5.1.7.3 [Page 5.1-4] In general, the uncertainty of the parameters used in the seismic analysis are understated, but the median values are considered to be conservative.

See Attachment 2 for response.

5.1.7.4 [Page 5.1-4] For seismic events, the core melt frequency may be low by a factor of 2 for IP-2 and by a factor of 8 for IP-3.

See Attachment 2 for response.

5.1.7.5 [Page 5.1-5] The tornado hazard curves are conservative, but the hurricane hazard curves are non-conservative.

See Attachment 1 for response.

5.1.7.6 [Page 5.1-5] Many statements in the wind fragility analysis are undocumented.

As is the case throughout the IPPSS, backup documentation for statements was included when deemed warranted. The decision as to when a statement is sufficiently important to justify including documentation in the text is necessarily subjective since the alternative is to substantially increase the volume of the already large documentation.

5.1.7.7 [Page 5.1-5] The major uncertainty in wind loading on an IP structure is due to the influence of nearby structures. The

analysis does not adequately represent the influence of adjacent structures.

See Attachment 1 for response.

5.1.7.8 [Page 5.1-5] The conversion of pressure to equivalent wind velocity ignores the shape factors of the buildings.

Building shape factors are used to determine the net wind loading on a structure. Such use in determining the factor of safety on loading is appropriate, but does not affect the safety factor on local parts of the structure. Since the shape factor was not applied to wind pressures when judging the margin of safety of a building to design wind loads, it would be inappropriate to take shape factors into consideration when making the final conversion of pressures to velocities, especially since the capacity of siding, girts, and other portions of a rectangular structure are generally independent of building shape factor.

5.1.7.9 [Page 5.1-5] The analysis presented in IPPSS for the loss-of-offsite power caused by wind is unconservative.

See Attachment 1 for response.

5.1.7.10 [Page 5.1-5] Based on the site visit by the review team, the possibility of either the turbine building or the superheater building failing due to wind and falling on the control building should be considered as well the latter falling on the diesel generator building.

See Attachment 1 for response.

5.1.7.11 [Page 5.1-5] The core melt frequency due to a hurricane at IP-2 is low by a factor of 20. The median hurricane hazard curve given in the IPPSS is too low, and the loss of offsite power analysis is unconservative.

As stated in Attachment 1, a revised analysis is in progress. However, we are confident that the additional analysis and results will show the reviewer's estimates of increased wind initiated core melt frequency to be grossly overstated.

5.1.7.12 [Page 5.1-5] The systems/components considered in the seismic and wind logic models seem to be reasonably complete.

No response required.

5.1.7.13 In the approach taken to evaluate the chances that external flooding would affect safety-related equipment, uncertainties were not adequately addressed, and only extreme events of low frequencies were considered.

As in several of the external events analyses, the approach taken in the external flooding analysis was to examine the magnitude and frequency of the worst events that could be postulated, using all the information that contributes to our state of knowledge, and, if that event could not lead to core damage, then no further analysis was necessary. That was the case during the IPPSS. Principal sources of information were contacted that could update our state of knowledge. Nothing was discovered to suggest that larger flood levels at the site were possible.

In search of new evidence regarding probable maximum flood at the Indian Point site as suggested in our meeting June 15, 1982, the U.S. Geological Survey (USGS) reference library and regional office were contacted (see references). It appears that no new study has been conducted on the Hudson River maximum flood by the USGS in the past decade, and no piece of evidence is available that would contradict the earlier findings as reported in the Indian Point external flooding analysis.

The maximum sustained water level at the site according to the attachment (Quirk, Lawler, and Matusky Engineers, letter dated January 21, 1972) is 14 feet above MSL. This elevation is calculated to be the result of standard project flood, standard project hurricane, and the failure of Ashokan Dam. Adding the wave effects to the flood level results in an instantaneous maximum elevation of 15.5 to 16 feet. However, neither the sustained nor the instantaneous maximum elevation pose a threat to the critical equipment and buildings.

The service water pumps which are located at Elevation 14' have their motors approximately 2 feet above that elevation. The control cabinet for service water is located in the switchgear room behind the turbine-generator building. In order to flood this room, either the turbine building with entrances at Elevation 15' must be flooded at a sustained level, or the water elevation has to reach 18 feet in close proximity to the other entrance to the switchgear room to be able to potentially cause flooding damage. However, both elevations are extremely unlikely in light of the calculated maximum sustained water level of 14 feet.

Therefore, since the currently assumed flood levels represent an upper bound flood from which no core damage would result, there is no basis within the present state of the art for considering additional uncertainty and for hypothesizing increasingly larger flood levels. This is consistent with the other external event analyses.

The following individuals were contacted to obtain updated statistics or analysis on the Hudson River maximum flood.

- Mr. William Sanders, Reference Librarian, USGS, Menlo Park, California. (No new report or document on the subject could be found. All the reports and documents were dated back to about 10 years ago.)

- Mr. Dan Eissler, USGS Regional Office. (He did not know of any new information or documents on Hudson River floods.)

5.1.7.14 While the analysis of internal flooding is not systematic, we agree that the effects on plant risk from internal flooding are small.

Consistent with the other events analyzed in the IPPSS, no detailed analysis was considered warranted if a potential plant initiating event could not be postulated.

5.1.7.15 [Page 5.1-5] The IPPSS fire analysis.

- appears to have identified all critical plant areas where a fire can cause an initiating event and, simultaneously, fail redundant safety systems.

No response required.

- has adopted the best available data base for estimating the frequency of fires in nuclear power plant areas.

No response required.

- appears to have identified all important safety system components and cabling which are located in critical plant fire areas.

No response required.

- reflects as-built plant conditions at the time the analysis was performed.

No response required.

- did not quantitatively assess the importance of a control room fire, even through an analytical basis for excluding the control room from analysis appears to be missing.

In a preliminary screening of the fire zones, the control room was not chosen as a critical location for the following reasons.

- The control room is directly above the cable spreading room and the instrumentation and controls on the control panels follow a configuration similar to that of the cable spreading room cable routing.

- The most critical area of the control room is the panel controlling the safety equipment. A fire in this cabinet would have the same impact on the plant as a fire at the center of the north wall. Similar to cable spreading room fire, the turbine-driven auxiliary

feedwater pump would remain operable in the case of a control room fire.

- The operators would detect any fire almost immediately and attempt to extinguish it in a short period of time.

- The power breakers of all the pumps can be closed locally at the switchgear, thus bypassing the control room. This can be achieved even in the case of a loss of DC power.

- All the valves are in their safe position. A hot short causing them to move to a nonsafe position is deemed to be of small conditional frequency given a control room fire.

- A fire in the electrical panel (the panel controlling the electric power system), in the worst case, would lead to a loss of AC power.

We believe that the analysis of fires in the control room would lead to a change of the low tail of frequency distribution of core damage due to fires in the cable spreading room.

5.1.7.16 [Page 5.1-6] The fire analysis assumes that fire damage occurs only through fire propagation within a fire plume. This may be non-conservative. In addition, significant operator recovery actions are allowed in a few fire situations, although confused operating conditions during a fires [sic] could hamper such actions. With more conservative assumptions in these two areas, the core melt frequency due to fire at Unit 2 can increase by a factor of 2 and at Unit 3 by a factor of 4.

See Attachment 3 for the discrepancy in the fire propagation modeling. The effect of changes in the human error contribution has an insignificant effect to the overall fire risk.

5.1.7.17 [Page 5.1-6] The analyses in the IPPSS concerning the transportation and storage of hazardous materials, turbine missiles, and aircraft accidents appear to be reasonable with their associated risks being negligible.

No response required.

5.1.7.18 [Page 5.1-6] Although external and internal events were considered separately in the external event logic models until containment systems were considered, the review substantiated the IPPSS hypothesis that combinations of such events are probabilistically small.

No response required.

5.1.8 ACCIDENT SEQUENCE ANALYSIS

5.1.8.1 [Page 5.1-6] In general, the IPPSS accident sequence analysis was difficult to follow because of: incorrect and/or incomplete references, nonmatching numerical results and unclear or inadequate description of events or the modeling of them.

We are pleased that the reviewers have pointed out incomplete references and nonmatching numbers. However, we believe that unclear descriptions have been kept to a minimum.

5.1.8.2 [Page 5.1-6] Reliance by the IPPSS on more representative fragility hazard curves would increase the seismic initiated, SE/2RW, sequence at IP-2 by a factor of two and at IP-3 by a factor of ten.

Refer to Attachment 2.

5.1.8.3 [Page 5.1-6] For the two dominant IP-2 fire scenarios listed in IPPSS Table 8.3-9, the SE/2RW scenario would increase by a factor of three and the SLF/8A scenario could not occur (and instead become part of the SE/2RW case) if the hot layer failure mechanism, described in Section 2.7.4, occurs. Similar observations hold for the fire scenarios of IP-3.

We disagree; refer to Attachment 3.

5.1.8.4 [Page 5.1-7] The assumption that loss of power to buses, 2A, 3A, and 6A at IP-2 leads to a seal LOCA is conservative because component cooling water has power as long as power to bus 5A is not lost. Thus, the frequency of scenario 4 in Table 8.3-9 is high by a factor of two. This also affects scenario 12.

The reviewers are correct in recognizing the conservatism embedded in the assumption for seal LOCA induced by loss of power to buses 2A, 3A, and 6A.

5.1.8.5 [Page 5.1-7] IPPSS may have underestimated the frequency of IP-2 scenario 5 in Table 8.3-9 by as much as a factor of 20 because of questionable assumptions made about the hurricane hazard at the site and the offsite power fragility.

Refer to Attachment 1.

5.1.8.6 [Page 5.1-5] Tornado initiated sequences at IP-2 appear to have been reasonably estimated.

No response required.

5.1.8.7 [Page 5.1-7] The failure of the operators to initiate switchover to recirculation is overestimated for the small LOCA sequences and underestimated for the large and medium LOCA sequences. This is true for both IP-2 and IP-3.

This question was addressed earlier in response to Sandia's last comment under Human Reliability Analysis (5.1.5.2).

5.1.8.8 [Page 5.1-7] The use of industry historical common cause pump failure data instead of the subjective IPPSS common cause value increases the contribution of system hardware failure in the internal accident sequences for both IP-2 and IP-3.

The statement is correct but generic industry data is inappropriate in many applications. However, use of generic data would not significantly affect plant risk.

5.1.8.9 [Page 5.1-7] The misuse of the failure criterion for diesel generator cooling by service water results in the overestimation of IP-2 scenarios 10 and 14 in Table 8.3-9 by greater than a factor of two. This is true as well for the equivalent IP-3 scenarios.

These scenarios have been requantified using the correct (less conservative) information and do not significantly affect core melt or public risk.

5.1.8.10 [Page 5.1-7] The sequence V analysis as presented in the IPPSS is wrong. At the request of the reviewers, the sequence for both IP-2 and IP-3 was reanalyzed. The results of the

revised analysis are not appreciably different than those presented in the IPPSS.

This question was addressed earlier in response to a Sandia comment under Estimation Methodology (5.1.6.2).

5.1.8.11 [Page 5.1-7] The use of more representative seismic hazard curves would increase the frequency of the IP-2 scenario 21 and IP-3 scenario 37 each by a factor of two.

Refer to Attachment 2.

5.1.8.12 [Page 5.1-7] In the IPPSS, some IP-3 sequence frequencies are higher than those of identical IP-2 sequences from what seems an overapplication of the data. We feel this is not justified.

The main issue behind this comment (see page 3.3.1-1 of Sandia's report for elaboration) is whether the experiential data from Indian Point 2 and 3 should be "pooled" or kept separate. The reviewers feel it should have been pooled; the IPPSS kept it separate (which is what they mean by overapplication of the data). Consider in particular Sandia's example of the safety injection pumps. If the data is pooled, the implicit assumption is that the failure rate of these pumps in Indian Point 2 is the same as it is in Indian Point 3. They support this assumption with the statement that "the systems (in the two units) appear similarly designed."

On the other hand, the IPPSS treated the data separately, which implies that the plants have their own failure rates independent of each other. The truth is, of course, somewhere in between; a proper handling of the situation would be a full two-stage Bayesian treatment when the data to support such becomes available. In the meantime, we feel that the assumption of independent failure rates is closer to the truth than that of identical rates. We feel that differences in maintenance and operating personnel and practices are major factors causing plant to plant variability of failure rates in spite of similarities in equipment design.

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COMPUTER APPLICATIONS

January 21, 1972

Job File: 115-8

Mr. John Inglima
Consolidated Edison Company of
New York, Inc.
4 Irving Place
New York, New York 10003

Subject: Flooding Conditions at Indian Point

Dear Mr. Inglima:

Pursuant to your request, we are submitting a summary description of the conservative assumptions which led to an estimated instantaneous water surface elevation some 6 to 12 inches higher than the design elevation of the 15.0 feet above mean sea level at the Indian Point Generating Station site. Such an elevation appears only in one set of the flooding conditions, as described in item 5 of our June 1971 report.*

As discussed with you in September 1971, the values listed in this table were revised to reflect our most recent estimates. For convenience, a copy of the revised table is attached to this letter.

We believe that interpretation of the meaning of this 6 to 12 inch increment should take into account the degree of conservatism adopted in the study. A brief discussion of the conservative assumptions employed in estimating the sustained elevation and wave runup values of item 5 of Table 1 is given below.

1. <u>Sustained Elevation of 14 feet above Mean Sea Level</u>

a. Simultaneous occurrence of the set of conditions considered in item 5 is probably the most conservative assumption utilized in this step. The resultant sustained elevation of 14 feet corresponds to a simultaneous occurrence of a Hudson River standard project flood, probable maximum precipitation over the Esopus Creek Basin which would result in Ashokan

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Quirk, Lawler & Matusky Engineers. "Supplemental Study of Flooding Conditions at Indian Point Nuclear Generating Unit No. 3," June 1971.

Letter to: Mr. John Inglima, Consolidated Edison Company of New York, Inc.

Date: January 21, 1972

Dam failure and standard project hurricane at New York Bay.

"Standard Project Hurricane" and "Standard Project Flood" are analogs and have been defined by the U.S. Army Corps of Engineers for a particular area (drainage basin) and season of year as a hypothetical hurricane (storm) intended to represent the most severe combination of hurricane (storm) parameters "reasonably characteristic" of a specified geographical region, excluding extremely rare conditions.*

Although the definition does not include reference to the probability, a low probability value can be expected, i.e., between 0.001 and 0.0001, or, in other words, the return period of such events is between 1,000 to 10,000 years.

However, item 5 of Table 1 assumes even more severe conditions, i.e., the overall storm resulting in the Hudson River standard project flood and the standard project hurricane at New York Bay combined with a probable maximum precipitation in the center of the overall storm. The storm center is assumed to cover the Esopus Creek drainage basin.

The probable maximum flood resulting from the probable maximum precipitation and critical runoff conditions in a drainage basin represents a boundary between possible and impossible floods, or in other words, the probability of occurrence of such a flood approaches zero.

Therefore, the chance of a simultaneous occurrence of all three events falls below the level of probability and into an interval of improbability.

Even if such a severe combination of conditions is assumed, it is highly unlikely that the center of the overall storm would be located over the Esopus Creek Basin in view of its location at the periphery of the Hudson River drainage area.

b. Computed sustained water surface elevation of 14.0 feet at Indian Point corresponds to simultaneous occurrence of maximum Hudson River flow of 705,000 cfs at Indian Point and maximum water surface elevation of 11.0 feet above mean sea level at the Battery. This is not necessarily the case, even if assumption of simultaneous occurrence of standard project hurricane at New York Bay and a storm resulting in the Hudson River standard project flood is accepted. Also, the assumption of constant flow of 750,000 cfs in the reach between Indian Point and the Battery is conservative.

c. The basic design requirement for earth dam stability during flooding conditions is the prevention of overtopping. In applying this principle

*U.S. Weather Report No. 33 (by H.E. Graham and D.E. Nunn) U.S. Weather Bureau, "Meteorological Characteristics of Maximum Probable Hurricane for Atlantic and Gulf Coasts."

to the Ashokan Reservoir dikes the more conservative assumption was made that the dikes would fail if the water surface reached an elevation of 597.0 above MSL in the West Basin, and/or an elevation of 594.0 feet above MSL in the East Basin, i.e., a height 1.0 foot above the crests of the concrete cores of the dikes.

Flood routing calculations (see Plate 5 of "Supplemental Study") show that even under the most severe flooding conditions resulting from probable maximum precipitation over Esopus Creek Basin, the water surface in the Ashokan Reservoir will remain safely below the crest of all dikes. Under these conditions, the maximum water surface elevation which could be reached in the West Basin is 604.8 feet above MSL, more than 5 feet below the crests of the West Basin dikes. The maximum elevation in the East Basin would be 598.8 feet above MSL, i.e., more than 7 feet and 3 feet below the crest of Middle and East Dikes, respectively.

The reasons for using the conservative assumption described above are as follows:

- The Ashokan Reservoir dam and dikes are old (the dam was built in 1930)

- There is no exact knowledge about the condition of the concrete cores. On the other hand, the minor discharges of leaking water through the dikes as measured under normal water surface elevations, indicate that the dike cores are probably in good condition.

- The capacity of the spillway connecting the two basins is of considerable importance in the correction of flood routing through the Ashokan Reservoir because the West Basin is not provided with any other outlet, and it controls 90% of the entire drainage area. Because the spillway would be submerged under PMF conditions and its discharge capacity has never been measured under such conditions, only an approximate equation was available to calculate this discharge.

2. <u>Wave Runup of 1.5 to 2.0 feet due to Local Wind</u>

a. Maximum wind speed was assumed to occur concurrently with maximum river surface elevation at Indian Point, although such a possibility is extremely remote.

b. The maximum wind speed inland reduction factor of 0.7 used in the Flooding Study has been derived by several investigators for a distance of five to ten miles from shore and for a relatively smooth shoreline.

Letter to: Mr. John Inglima, Consolidated Edison Company of New York, Inc.

Date: January 21, 1972

Considering the distance between the Indian Point Site and the Atlantic Coast shoreline of about 40 miles and the relatively rough configuration of the Hudosn River Valley, the inland factor could logically be reduced. However, such a reduction of maximum wind speed would require the detailed analysis of a significant amount of wind speed data as measured in the geographical region.

If, for example, the maximum wind speed were reduced to about 50 mph, then the significant wave runup would be approximately 1 to 1.3 feet and the maximum water surface elevation at Indian Point would become 15.0 to 15.3 feet above MSL.

The set of flooding conditions as delineated above, is the assumption which led to the highest water surface elevation at Indian Point. However, this set of conditions was included in our study at the request of the U.S. Atomic Energy Commission. Use of less conservative assumptions in the analyses would require additional work with extreme difficulty in obtaining acceptable data for such assumptions.

The six sets of conditions as summarized in Table 1 should be considered in the light of attempts to estimate the water surface elevations under very improbable and severe conditions. In most cases, the analyses show that the maximum possible water surface elevation would be less than 15.0 feet above mean sea level. In only one case was the surface elevation six to 12 inches above the design elevation of 15 feet above MSL. This six to 12 inch range is within the limits of the accuracy of the analyses.

Very Truly Yours,

Karim A. Abood

Karim A. Abood
Project Manager

KAA:gsf
enc. 1

WATER SURFACE ELEVATIONS AT INDIAN POINT RESULTING FROM STATED FLOW AND ELEVATION CONDITIONS

Component Flow at Indian Point	Elevation at the Battery (MSL Datum)	Flow Indian Point (millions of cfs)	Sustained elevation at Indian Point (MSL Datum)	Significant wave runup (ft)	Instantaneous Maximum Elevation (MSL Datum)
1. Probable maximum flood	Mean Sea Level (0.00)	1.100	12.7	+1.4	14.1
2. Probable maximum flood & tidal flow	Mean High Water (+2.2)	1.014+	12.7	+1.4	14.1
	Mean Low Water (-2.2)	1.165+	12.7	+1.4	14.1
	Spring High Water (+2.7)	1.179+	12.7	+1.4	14.1
	Spring Low Water (-2.7)	0.998+	12.7	+1.4	14.1
3. [illegible]dard Project [illegible] & Ashokan [illegible] failure	Mean Sea Level (0.00)	0.705	7.2	+1.4++	8.6
4. Standard Project flood	Standard Project hurricane (+11.0)	0.550	13.0	1.5-2.0*	14.5-15.0
5. Standard Project flood & Ashokan Dam failure	Standard project hurricane (+11.0)	0.705	14.0	1.5-2.0*	15.5-16.0
6. Probable Maximum hurricane & Spring high tide	Probable maximum hurricane (+17.5)	---	12.4	2.0-2.5**	14.4-14.9

NOTES:

* Standard project hurricane wave runup determined for:
- forward speed of hurricane ..34 knots
- maximum speed of hurricane (inland factor 0.7) ..75 MPH
- duration of maximum wind speed ..0.13 hrs

\+ Flow corresponds to reach of the Hudson River affected by tidal variation under probable maximum flood conditions. [illegible] reach extends from the Battery to the Tappan Zee [illegible]e, about mile point 27. Actual flow at Indian Point, [illegible] 16 miles above the Tappan Zee Bridge is 1.100 million cfs.

** Probable maximum hurricane wave runup determined for:
- forward speed of hurricane ..34 knots
- maximum speed at indian Point (inland factor 0.7) ..90 MPH
- duration of maximum wind speed ..0.13 hrs

++ Wave runup assumed approximately the same as for PMF conditions.