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Letter Report
on
Review and Evaluation
of the
Indian Point Probabilistic Safety Study

August 25, 1982

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Prepared for

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Indian Point Letter Report

1. Introduction

Sandia National Laboratories has performed a limited review of the system analysis and external events analysis of the Indian Point Probabilistic Safety Study[1] (IPPSS)for the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission,. The review has been conducted over a three and one-half month period, by Sandia personnel with contractor support. To date, approximately 2C man-months effort has been expended in the review. The results provided in this report should be considered preliminary pending NRC review. Following that review a final letter report will be released.

The review has focused on the plant analysis and external events analysis of the Indian Point study. Each major topic area of the plant analysis portion of the study was reviewed: initiating events, event trees, success criteria, fault trees, human reliability analysis, component data, and uncertainty. The treatment of external events, including seismic, fires, floods, missiles, wind, transportation of hazardous materials, and aircraft crashes, was also reviewed. Not every topic was reviewed in detail. Emphasis of our review was on those portions of the analysis which appeared most important to the results of the Indian Point study.

In addition to each topical area, the important accident sequences from the study were reviewed in detail. The sequences dominating risk were reviewed in detail as well as sequences important to the core melt probability but contributing little to risk due to the low consequences anticipated for these accidents. The intent of the sequence review was to evaluate the analysis of the Indian Point study and to evaluate the changes in the estimated frequencies of the sequences which could arise from differences in assumptions and the treatment of data.

Several issues and assumptions were evaluated in addition to the sequences. The issues were chosen as a result of interest on the part of NRC or because of their having been important in other risk assessments. Several of these issues, such as feed and bleed capability and interactions between core melt and containment systems, are issues for which somewhat controversial assumptions must be made which may differ between analysts. Other issues, such as anticipated transients without scram, are generic, unresolved safety issues. Still others, such as the treatment of reactor

[1]Indian Point Probabilistic Safety Study, Power Authority of the State of New York, Consolidated Edison Company of New York, Inc., Spring 1982.

coolant pump seal LOCAs, arose because of peculiarities of the Indian Point plants. These issues were generally treated in the manner of a sensitivity study. Assumptions were varied to see what the effects on the results could be. Often, this took the form of a bounding calculation.

It should be noted that the primary emphasis of the review was to search for <u>significant</u> errors and uncertainties in the IPPSS. We therefore did not keep close account of all small errors and uncertainties (e.g., those that affect the core melt frequency or risk by approximately less than a factor of two).

The preliminary results of our review are presented in the following sections. The review of plant analysis and external events topics is presented in Section 2. Section 3 presents the review of selected accident sequences. Section 4 details the review of selected issues. Section 5 summarizes the principal findings and presents estimates of plant damage state frequencies for use in containment and consequence calculations.

2. Areas of Review

The IPPSS, as any PRA, is composed of several interrelated tasks. A review of a PRA is not complete unless the information and analysis which comprises each task is examined. The IPPSS PRA tasks are depicted in Figure 2-1. Also shown there are the letter report sections which summarize our review of a task. As can be seen, this letter report does not represent a complete review since several of the PRA tasks were not examined. For the most part, these omissions are in the containment and consequence analysis areas. A review of these areas will be enfolded into future revisions of this report. The only task which we intend not to review in depth for the final report is the first -- "initial information collection." Our review therefore generally assumes that the IPPSS has collected accurate Indian Point design and operations information; e.g., correct piping and instrumentation layouts, etc.

2.1 Initiating Events

The initiating events covered in the IPPSS seem to be relatively complete compared to those addressed in previous PRAs. The initiating event categories analyzed were identical for both Indian Point units. IPPSS Table 1.5.1-23 summarizes the initiating events considered and is reproduced here for reference. The treatment of these initiating events is discussed in other sections of this review.

Comparisons were made to other PRAs, EPRI NP-801, and to an NRC list of concerns about potentially omitted initiating events. In addition, several initiating events were identified by NRC as being of particular interest. These are discussed below.

1) Excess Letdown or Decreased Charging

The result of this potential initiating event is lowering the reactor coolant inventory without detection to a level that would require reactor trip and mitigation. Although not addressed specifically in the IPPSS, this event would be included in the EPRI NP-801 data used to quantify the reactor trip event.

2) Insufficient Letdown or Increased Charging

This potential initiating event would cause RCS overpressure and thus falls under subcategory 12 (IPPSS Table 1.5.1-23) and is included in the EPRI NP-801 data used to quantify the initiating event.

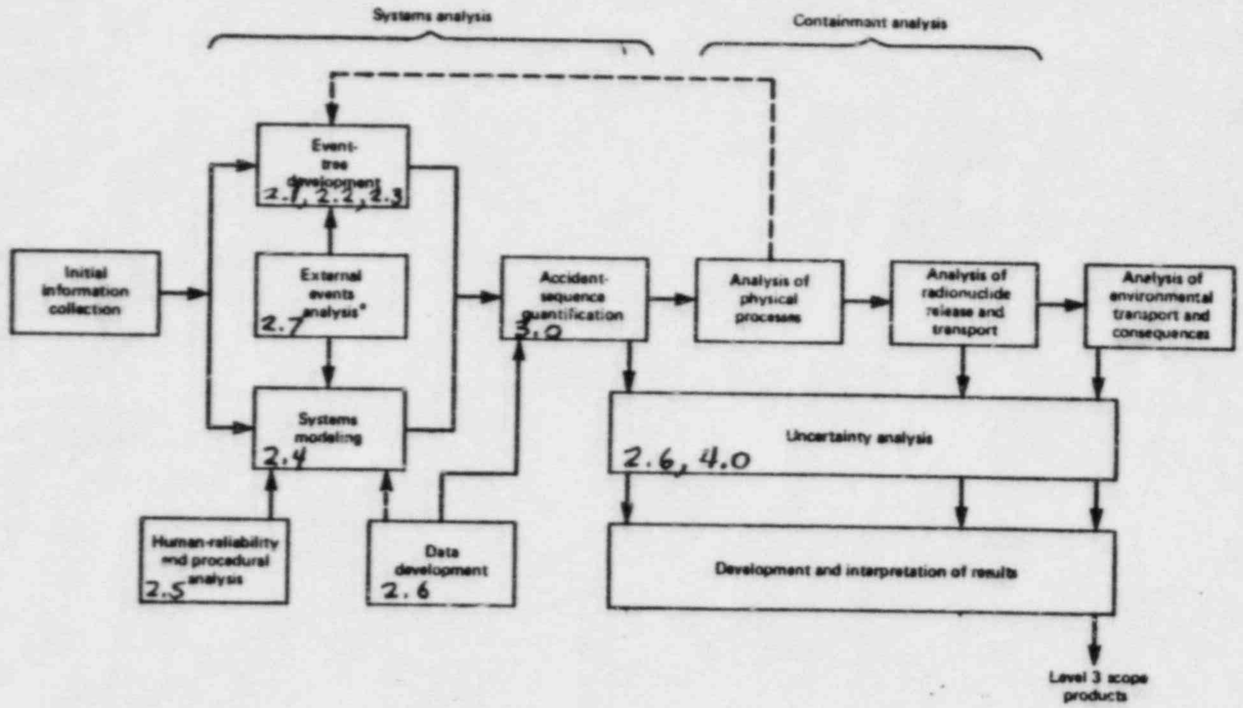

Figure 2-1. Risk-assessment procedure.

TABLE 1.5.1-23

INDIAN POINT 2 INITIATING EVENT SUBCATEGORIES

1. Large Loss of Coolant Accidents (LOCAs)

2. Medium LOCAs

3. Small LOCAs

    a. Pressurizer relief or safety valve opening
    b. Miscellaneous small LOCAs

4. Steam Generator Tube Rupture

5. Steam Pipe Rupture Inside the Containment

6. Steam Pipe Rupture Outside the Containment

7. Loss of Feedwater Flow

    a. Loss/reduction of feedwater flow in one steam generator
    b. Loss of feedwater flow in all steam generators
    c. Feedwater flow instability--operator error
    d. Feedwater flow instability--mechanical causes
    e. Loss of one condensate pump
    f. Loss of all condensate pumps
    g. Condenser leakage
    h. Miscellaneous secondary leakage

8. Full or Partial Closure of One Main Steam Isolation Valve (MSIV)

9. Loss of Primary Flow

    a. Loss of primary flow in one loop
    b. Loss of primary flow in all loops

10. Core Power Increase

    a. Uncontrolled rod withdrawal
    b. Boron dilution--chemical and volume control system malfunction
    c. Cold water addition

TABLE 1.5.1-23 (continued)

INDIAN POINT 2 INITIATING EVENT SUBCATEGORIES

11. Turbine Trip

    a. Turbine trip

        1. Closure of all main steam isolation valves
        2. Increase in feedwater flow in one steam generator
        3. Loss of condenser vacuum
        4. Loss of circulating water
        5. Throttle valve closure/electro-hydraulic control problems
        6. Generator trip or generator caused faults
        7. Increase in feedwater flow in all steam generators

    b. Turbine trip due to loss of offsite power
    c. Turbine trip due to loss of service water

12. Reactor Trip

    a. Reactor trip

        1. Control rod drive mechanism problems and/or rod drop
        2. High or low pressurizer pressure
        3. Spurious automatic trip--no transient condition
        4. Automatic/manual trip--operator error
        5. Manual trip due to false signal
        6. Spurious trip--cause unknown
        7. Primary system pressure, temperature, power imbalance
        8. Loss of power to necessary plant systems
        9. Spurious safety injection activation

    b. Reactor trip due to loss of component cooling water

3) Pressurized Thermal Shock

This is a safety issue not addressed by the IPPSS or any of the current or past PRAs. It is a complex issue which requires very detailed plant specific probabilistic, thermohydraulic, and fracture mechanics analysis. Due to the time limitations placed on this review, we were not able to evaluate this initiating event.

4) Failure of the Pressurizer Sprays or Heaters

This initiating event results in loss of RCS pressure control and thus falls under subcategory 12 (IPPSS Table 1.5.1-23) and is included in the EPRI NP-801 data used to quantify the initiating event.

5) Inadvertent Containment Spray Operation

This initiating event was not treated explicitly in the IPPSS or in previous PRAs. Existing data from EPRI NP-801 indicates this a low probability initiating event and it does not cause more stressing conditions than higher probability initiating events considered in the IPPSS. Therefore it is not considered a significant contributor to risk.

The above temperatures were based on the following assumptions:

- all equipment in the auxiliary building operational except ventilation,

- no credit taken for natural circulation through ducts, and

- gradient calculated between building structure and ambient temperature at H = 1.5 Btu/hr ^oF ft^2.

Based on this analysis, United Engineers predicts no safety related equipment failures within the first 24 hours following a loss of auxiliary building ventilation.

11) Reactor Coolant Pump Seal Failure

The RCP seal failure initiating event should be considered a small LOCA. However, it is not included in the data base. While it is true they were not included, the small LOCA frequencies of approximately .02 quoted in the report for each Indian Point unit is a reasonable estimate. The NRC has conducted a study of RCP rupture LOCAs which suggests their frequency to be approximately .02 (NRC memo from Thomas Morley to Darrel Eisenhut, Subject: Reactor Coolant Pump/Seal Failure, no date). Conceivably, the Indian Point small LOCA frequencies could be .04. However, upon review of the data comprising the IPPSS small LOCA frequency, it was noted that many of the small LOCA events involved stuck open pressurizer PORVs. It is generally known that some of these events were recovered by the operators in a few minutes via closure of the PORV block valves. The IPPSS did not consider recovery and thus probably overpredicted the frequency of PORV LOCAs. It is felt that this overprediction would tend to cancel the underprediction of RCP seal failure and thus the small LOCA frequency estimate of .02 is reasonable.

12) Loss of Component Cooling Water Due to a Pipe Break

This potential initiating event could conceivably lead to core melt unless judicious operator recovery actions are performed within about an hour. Assuming no operator recovery, a large pipe break in the component cooling system would cause a reactor trip, could eventually cause a reactor coolant pump seal LOCA, and failure of the pumps which provide makeup to the reactor coolant system. It should be noted that the IPPSS analyzed a "loss of pump flow" induced loss of component cooling water initiating event. However, the IPPSS did not analyze one induced by a pipe break. The system responses are quite different for the two cases and thus this is considered as a potentially important omission.

In conclusion, review of the NRC list of potential IPPSS initiating event omissions has indicated that pressurized thermal shock and loss of component cooling water due to a pipe break appear to be the only potentially significant events omitted in the IPPSS. The rest are implicitly or explicitly included in the IPPSS existing initiating events or are judged to be insignificant.

The loss of component cooling water due to a pipe break is evaluated in Section 4.6 of this review. As stated earlier, an evaluation of pressurized thermal shock does not appear in this review.

It should be noted that seven external initiating events (seismic, fire, flood, wind, aircraft accidents, transportation and hazard materials, and turbine missiles) were considered, which is more than most PRAs have attempted. The external event review appears in Section 2.7.

Initiating Event Quantification

Estimated initiating event frequencies are expected to vary from plant to plant depending on the plant characteristics, design, and its specific data base. The IPPSS initiating event data were compared to the data used in the Arkansas Nuclear One (ANO) IREP analysis. (The reason for choosing ANO is because it is a recently completed NRC-sponsored PRA which they support.) The purpose of the comparison was to look for unusual differences that might indicate a potential error in judgment or calculation. The mean values from the IPPSS are:

Initiating Event Category	Occurrences/Year IP2	IP3
1. Large LOCA	1.9×10^{-3}	2.2×10^{-3}
2. Medium LOCA	1.9×10^{-3}	2.2×10^{-3}
3. Small LOCA	1.9×10^{-2}	2×10^{-2}
4. Steam Generator Tube Rupture	2.7×10^{-2}	3.4×10^{-2}
5. Steam Break Inside Containment	1.9×10^{-3}	2.2×10^{-3}
6. Steam Break Outside Containment	1.9×10^{-3}	2.2×10^{-3}
7. Loss of Main Feedwater	6.7	3.8
8. Trip of One MSIV	1.3	9×10^{-2}
9. Loss of RCS Flow	1.4×10^{-1}	1.7×10^{-1}
10. Core Power Excursion	2.2×10^{-2}	2.6×10^{-2}
11a. Turbine Trip	7.3	2.7
11b. Turbine Trip--Loss of Offsite Power	1.8×10^{-1}	2.7×10^{-1}
11c. Turbine Trip--Loss of Service Water	1.9×10^{-3}	2.2×10^{-3}
12a. Reactor Trip	6.8	2.9
12b. Reactor Trip--Loss of Component Cooling	1.9×10^{-3}	2.2×10^{-3}
V. Interfacing System LOCA	4.7×10^{-7}	4.6×10^{-7}

The ANO PRA utilized WASH-1400 data for breaks greater than 2". For breaks less than 2" WASH-1400 data was added to the 2×10^{-2} reactor coolant pump seal rupture data discussed in 11 above. The following compares ANO, WASH-1400, and IPPSS LOCA frequency data:

		IP2	IP3	ANO	WASH-1400
1. Large LOCA	6"	$1.9x10^{-3}$	$2.1x10^{-3}$	$1x10^{-4}$	$1.0x10^{-4}$
2. Medium LOCA	2"-6"	$1.9x10^{-3}$	$2.1x10^{-3}$	$3x10^{-4}$	$3.0x10^{-4}$
3. Small LOCA	2"	$1.8x10^{-2}$	$2x10^{-2}$	$2.1x10^{-2}$	$1.0x10^{-3}$
4. Interfacing Systems LOCA		$4.7x10^{-7}$	$4.6x10^{-7}$	$< 10^{-6}$	$4x10^{-6}$

The IPPSS frequencies are larger on two of the three LOCAs and approximately equal for the small LOCA. (The larger estimates for large and medium LOCA are due to the use of Bayesian methodology in the IPPSS--see Section 2.6 of this review). The interfacing systems LOCA has a smaller estimate in the IPPSS because of more frequent testing of the low pressure injection check valves than the Surry Plant in WASH-1400.

Transients are subdivided differently at ANO but five are directly related.

	IP2	IP3	ANO
7. Loss of Main Feedwater	6.7	3.8	1.0
11b. Turbine Trip-- Loss of Offsite Power	$1.8x10^{-1}$	$2.7x10^{-1}$	$3.2x10^{-1}$
11c. Turbine Trip-- Loss of Service Water	$1.9x10^{-3}$	$2.2x10^{-3}$	$2.6x10^{-3}$
11a. Turbine Trip--	7.3	2.8	
12a. Reactor Trip--	6.8 } 14.1	2.9 } 5.7	7.1

The IPPSS transient initiating event frequencies appear reasonable; the differences are the result of the influence of plant specific data. (The other IPPSS initiating events were not explicitly analyzed at ANO because they were either a) not applicable, b) were not identified to be risk significant, or c) grouped with other transients.)

The reactor vessel rupture LOCA (R), is not considered mitigable and thus leads to core melt by itself. The IPPSS concluded that the frequency of such an event is small compared to other events leading to the same plant damage state; e.g., large LOCA followed by failure of low pressure injection. This may be an erroneous conclusion since the most likely vessel rupture sequence may be caused by pressurized thermal shock. As stated earlier, pressurized thermal shock was not analyzed in the IPPSS.

In summary, none of these frequencies appear to be out of line with what would be expected from experience and from what is reported in the ANO PRA.

2.2 Event Trees

The IPPSS constructed 13 event trees to model the plant system response to the initiating internal events discussed in Section 2.1. We reviewed these trees for validity. During the review, several questions were generated which could not be answered by information or analysis presented in the text. These questions were, for the most part, answered during a meeting held in June 1982 between Sandia Laboratories and IPPSS personnel. The findings are of two types. General findings are those that apply to all or several of the event trees. Specific findings are those that apply to a particular event tree. These findings and the impact they may have on the IPPSS results will now be discussed.

2.2.1 General Event Tree Findings

Containment Spray System Analysis

There are two containment spray systems installed at an Indian Point unit. The containment spray injection system (CSIS) consists of two pump trains which take suction from the refueling water storage tank (RWST). Upon depletion of the RWST, the CSIS pumps are shut down. During the recirculation phase, the containment spray recirculation system (CSRS) is utilized. The CSRS is a two-train system which utilizes the same pumps as the low pressure recirculation system (LPRS). A portion of the LPRS flow is diverted to the CSRS spray headers. During recirculation, the LPRS pumps take suction from the containment sump.

Though not explicitly stated in the IPPSS, no credit was given on the event trees for operation of the CSRS. Referring to event tree 2 (IPPSS Figure 1.3.4.2-1, event tree 2), for example, it can be seen that on sequences 43 and 46, the CSRS is defined to be operating yet the plant damage state (AEF and AE, respectively) implies that sprays are not operating. This is a conservatism adopted in the Indian Point analysis and may be justified for the following reasons: 1) In the vast majority of core melt sequences the PRA analyzed, the LPRS is unavailable. Since the LPRS and CSRS share most of the same equipment, the CSRS would most likely also be unavailable; 2) during a core melt accident, the LPRS/CSRS pumps may fail since their sump water supply could be clogged with core melt debris.

The CSIS, on the other hand, is given more credit on the event trees than may be justified. Upon close examination of event tree sequence plant damage states (e.g., sequence 2, event tree 2) and the CSIS event definitions it is noted the IPPSS assumes the CSIS is available during the recirculation phase if it was successful during the injection phase. In order for the CSIS to be available during recirculation, the RWST must be refilled by the operators. We

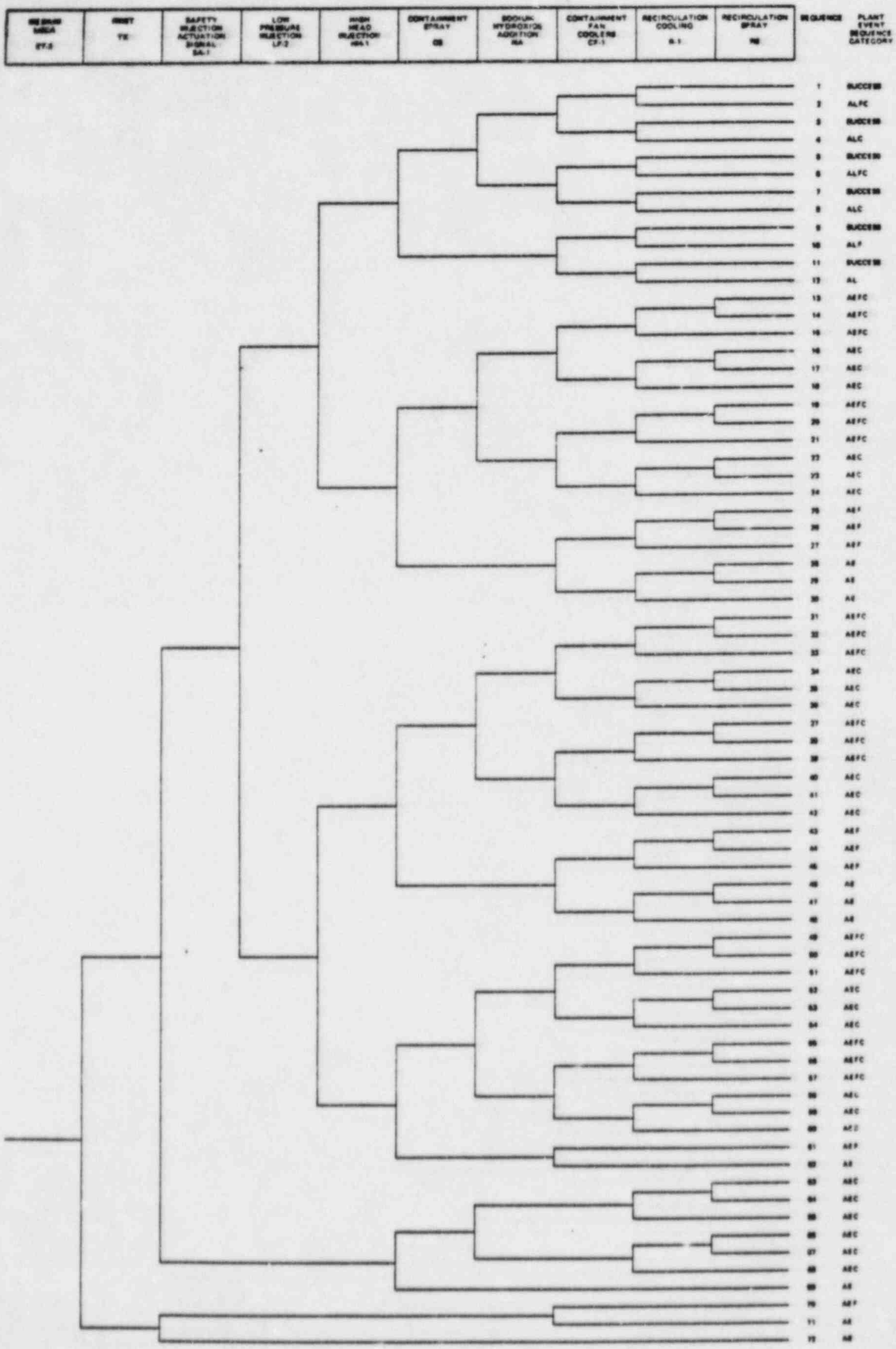

Figure 1.3.4.2-1. Medium LOCA - Event Tree 2

1.3-165

question the validity of giving credit for the CSIS during recirculation since we find no mention of refilling the RWST in the Indian Point LOCA emergency procedures.

If one assumes that the CSIS will not be available during the recirculation phase, all core melts that are initiated during the recirculation phase would not have sprays available to mitigate the consequences of the accident. This implies that plant damage states characterized by C (spray injection operating) and L (core melt initiated in the recirculation phase) are not possible. Thus, IPPSS damage states SLFC, ALFC, SLC, and ALC become SLF, ALF, SL, and AL, respectively. The impact this finding has on the plant damage state frequencies is presented in Section 4.2.

Core Melt/Safety System Interactions

The interdependencies incorporated into the Indian Point event trees imply that the containment systems (i.e., the spray system and fan cooler system) may be utilized during a core melt accident. This is an important assumption since the Indian Point analysis predicts that the operation of either of these systems can significantly reduce the risk associated with a core melt accident. This topic is discussed more fully in Section 4.2.

Sodium Hydroxide Addition

All event trees model the additions of sodium hydroxide to the containment spray water. This was modeled because it was thought to enhance the radioactive material scrubbing capability of the spray water during a core melt accident. Discussions with IPPSS personnel revealed that analysis performed late in the study indicated that sodium hydroxide addition had a negligible effect on the assessment of plant damage states and release categories. All event trees could therefore be simplified by removal of the sodium hydroxide addition event. This is consistent with the findings of WASH-1400 with respect to sodium hydroxide addition.

Main Feedwater System

The Indian Point study assumed that the main feedwater system was unavailable for purposes of removing post shutdown decay heat following all internal and external initiating events analyzed.

Other PRAs of PWRs have typically assessed the main feedwater system to be unavailable following a LOCA, a loss of offsite power, or an ATWS. For the remaining internal initiating events, these PRAs assessed that for other initiating events caused by power or cooling water failures the unavailability of the main feedwater system was within the range of .1 to .01.

The assumption made in the IPPSS that the main feedwater system is always unavailable should not appreciably affect either the assessed core melt frequency or plant risk. Review of the Indian Point dominant accident sequences indicates that the core melt frequency and plant risk are dominated by external events and LOCAs and losses of AC power. Not giving credit for main feedwater during these accidents is consistent with other PRAs as mentioned in the previous paragraph.

Operator Actions Performed During Event Tree Accident Sequences

In response to the accidents modeled on the IPPSS event trees, Indian Point operators are expected to follow one or several emergency procedures. These procedures outline the actions the operators perform to contend with the accident. If the actions are performed correctly, core damage can be avoided. The operator actions outlined in the Indian Point procedures were compared with the IPPSS analysis of operator actions required to prevent core damage. In some cases, we found that the IPPSS modeled operator actions which were not outlined in the procedures. The most notable example of this is "feed and bleed" core cooling. The IPPSS assigned a high probability of success of this core cooling method, yet no feed and bleed procedures exist at Indian Point 2 and very limited ones exist at Indian Point 3. The impact that not giving credit for feed and bleed core cooling has on the plant damage state frequencies is presented in Section 4.3.

Based on discussions we had with IPPSS personnel, we discovered that the IPPSS did not model all operator actions which could affect the course of an event tree accident sequence; i.e., they did not calculate a probability of human error, but rather assumed the operator performed a task with a negligible failure probability. An example was mentioned in the discussion of the containment spray injection system analysis. As discussed there, the IPPSS assumes the operator will refill the RWST to allow continued operation of the CSIS. We could not find an analysis in the PRA which assesses the probability of operator error in performing this task. If the probability of human error is close to unity, the effect on the plant damage states would be the same as discussed previously.

Core Melts Caused By Containment Overpressure Failure

The Indian Point event trees do not model core melts caused by containment overpressure failure. These sequences have been shown to be important in other PRAs (e.g., the S_2C sequence in WASH-1400). Before discussing how this potential omission affects the Indian Point analysis, we will discuss how other PRAs have typically modeled core melts caused by containment overpressure failure.

The S_2C type of sequence is a LOCA followed by failure of the containment heat removal system. Failure of containment heat removal causes the containment to overpressurize and fail within several hours due to LOCA steam evolution. Up until the point of containment failure, the core is protected by successful operation of the emergency core cooling system during the injection and recirculation phase. Just prior to containment failure, the water in the sump is greater than 300°F, but is subcooled relative to the containment pressure (>100 psig). When the containment fails, it undergoes a rapid depressurization to atmospheric pressure and thus causes the sump water to boil vigorously. This boiling is postulated to cause cavitation failure of the emergency core cooling pumps which are at the time taking suction from the sump. Failure of these pumps causes core melt.

The above sequence discussion assumes a core melt can be prevented prior to containment failure by pumping water hotter than 300°F to the core. This is not the assumption made in the IPPSS. The IPPSS system models assume that during the recirculation phase the water delivered to the core must be cooled by component cooling water or a core melt will ensue. The requirement for component cooling water essentially assures the core will melt prior to a containment overpressure failure. The scenario discussed in the previous paragraph is therefore not possible by IPPSS models.

The IPPSS treatment is potentially nonconservative in terms of predicting risk since, in their models, a core melt will occur before containment failure. This allows a certain amount of radioactive material to plate out within containment <u>prior</u> to failure. In the S_2C type sequence, however, core melt occurs <u>after</u> containment failure and thus the release of radioactive material would be greater.

We have assessed the effect that this potential nonconservatism has on the risk predicted in the IPPSS and found it to be negligible. A review of the Indian Point core cooling and containment heat removal systems indicated that it is almost completely assured that if core cooling during the recirculation phase is provided, so also will containment heat removal. This is because the core cooling and one of the containment heat removal systems share most of the same equipment (i.e., pumps and support systems). Because of this dependence, the probability of having core cooling and not containment heat removal is negligible.

<u>Transient Induced Pressurizer Safety Valve Demands</u>

The IPPSS event trees do not model the demand of the pressurizer safety valves in response to a transient. This raised a concern that the study may have missed some important accident

Event Tree 11c - Turbine Trip Due to a Loss of Service Water and Event Tree 12b - Reactor Trip Due to a Loss of Component Cooling Water

The IPPSS used the turbine trip and reactor trip event trees to model the plant response to a loss of service water and loss of component cooling water initiating event respectively. These event trees do not adequately model the plant response to these initiating events for the following reasons:

a) the trees to not allow for a reactor coolant pump (RCP) seal LOCA to occur following a sustained loss of component cooling or service water,

b) the systems which respond to a seal LOCA are not adequately modeled, and

c) station blackout caused by a loss of service water and a loss of offsite power is not modeled.

If a loss of component cooling occurs, the RCP seals will lose cooling within approximately 15 minutes due to failure of the charging pumps and cooling to the thermal barrier heat exchanger. The IPPSS predicts a 1200 gpm seal LOCA will occur approximately 30 minutes following a loss of seal cooling. If the safety injection pumps subsequently fail, a core melt would ensue.

Service water cools component cooling water via two heat exchangers. If service water to the heat exchangers fail, the component cooling system would heat up at approximately 5^{o}F/hour (Reference: Con Ed FSAR, AEC question 6.5). If service water to the heat exchangers is not restored within several hours, RCP seal cooling and the safety injection pumps could fail. This could lead to a LOCA followed by core melt.

Service water also cools the diesel generators. If service water fails, followed by a loss of offsite power, the diesels would fail in a short time. This would result in a station blackout. If AC power is not restored within approximately an hour, a seal LOCA could occur followed by core melt. If AC power is not restored within approximately 3 hours, a containment overpressure failure leading to a 2RW radioactive material release could occur.

An abbreviated analyses was performed to determine the most likely sequences the IPPSS failed to model. We found the sequences frequency estimates to be small compared to other sequences which appear in the same plant damage state.

Event Tree 13--ATWS

The event tree appearing in the IPPSS does not represent the as built plant response to ATWS events. At the time the IPPSS was conducted, Con Ed and PASNY committed to perform plant modifications upon the recommendations of NUREG-0460 to reduce ATWS risk. Event tree 13 represents the plant response after the modifications are in place. Recently, Con Ed and PASNY have decided not to implement ATWS modifications and thus the event tree must be significantly modified. Because of this, IPPSS personnel are constructing a new event tree. It will appear in a soon to be released supplement to the IPPSS. The impact on risk of not installing ATWS modifications at Indian Point is investigated in Section 4.4.

2.3 Mitigating Systems Success Criteria

In response to LOCA and transient initiating events, various Indian Point core cooling and containment systems are called upon to bring the plant to a safe shutdown condition. If core cooling is unsuccessful and a core melt ensues, the containment systems may still be able to reduce the consequences of the accident by maintaining the containment boundary and thus isolating the core melt from the environment. The combinations of plant systems required to cool the core and maintain the containment boundary constitute the Indian Point mitigating system success criteria. We have reviewed the validity of the success criteria employed in the IPPSS. We have judged the success criteria to be consistent with criteria employed in PRAs of similar plants.

Table 2.3-1 summarizes the LOCA and transient success criteria employed in the IPPSS.

The IPPSS did not employ the containment overpressure protection success criteria stated in the FSAR. The FSAR criteria is 2/2 sprays OR 5/5 fans OR 1/2 sprays and 3/5 fans. The FSAR criteria applies to keeping containment pressure below the design pressure. The IPPSS criteria apparently applies to keeping pressure below failure pressure, i.e., greater than double design pressure. We found no reference in the IPPSS for the criteria used. However, the IPPSS criteria employed is supported by analysis of the Oconee containment systems as part of the Reactor Safety Study Methodology Applications Program. In that study it was shown that one fan or one spray, which had similar heat removal capabilities as Indian Point, adequately maintained the containment pressure within acceptable limits. Since both Oconee and Indian Point have similar MW ratings and containment volumes, it is judged that the IPPSS criteria is reasonable.

We could not find in the Indian Point FSARs an explicit statement of the core cooling success criteria in response to the full range of potential LOCA break sizes and transient initiating events. The IPPSS apparently made use of some Westinghouse documents and the FSAR in establishing the criteria employed in the report. (The FSAR "roughly" defined a criteria which was similar to that used in the IPPSS.) The IPPSS gave credit for "feed and bleed" core cooling during transients and small LOCAs following failure of the auxiliary feedwater system. Feed and bleed cooling is still an open question (see Section 4.3), but recent tests at the LOFT facility have suggested that it is a viable core cooling option. Though we could not validate the entire core cooling success criteria employed in the IPPS, it is our opinion that it is reasonable since it is similar to that used in other PRAs with which we are familiar.

Table 2.3-1 IPPSS LOCA and Transient Mitigating System Success Criteria

LOCA SIZE	Emergency Core Cooling Early (RWST)	Emergency Core Cooling Late (SUMP)	Containment Overpressure Protection	Radioactivity Removal
0-2"	1/3 Safety Injection Pumps (SI) and 1/3 Auxiliary Feedwater Pumps (AFWS) OR 1/3 SI and 2/2 PORVs	1/3 SI and 1/2 RHR Pumps OR 1/3 SI and 1/2 Recirc. Pumps	1/2 Containment Spray Pumps OR 3/5 Containment Fans	1/2 Containment Spray Pumps
2-6"	2/3 SI and 1/2 RHR Pumps	2/3 SI and 1/2 RHR OR 2/3 SI and 1/2 Recirc. Pumps	Same	Same
>6"	3/4 Accumulators and 1/2 RHR Pumps	1/2 Recirc. Pumps OR 1/2 RHR Pumps	Same	Same
Steam Generator Tube Rupture	1/3 AFWS and RCS Depressurization OR 1/3 SI and RCS Depressurization OR 1/3 SI and 2/3 AFWS	1/3 SI and 1/2 RHR OR 1/3 SI and 1/3 Recirc. Pumps	Same	Same

TRANSIENTS

Emergency Core Cooling Early (Secondary or RWST)	Emergency Core Cooling Late (Secondary or SUMP)	Containment Overpressure Protection	Radioactivity Removal
1/3 AFWS OR 1/3 SI and 2/2 PORVs	1/3 AFWS OR 1/3 SI and 1/2 RHR OR 1/3 SI and 1/2 Recirc.	Same	Same

2.4 Review of IPPSS Fault Trees for Indian Point 2 and 3

The system fault trees presented in IPPSS Sections 1.5 and 1.6 (for Indian Point 2 and 3, respectively) were reviewed for accuracy and completeness. The findings of this review are presented in this section of the report.

2.4.1 Fault Trees of Indian Point 2

Section 1.5 of the IPPSS presents the systems analyses for the Indian Point 2 reactor. The review of these analyses is presented below. Unless otherwise noted, system failure probabilities cited herein are for the case of all power available, which is of primary concern except for a few systems. By the nature of the review, only those areas of disagreement are discussed.

2.4.1.1 IP-2 Emergency Electric Power System Fault Tree

The emergency electric power system fault tree for IP-2 was extensively reviewed, in particular its intra-actions among its AC, DC, and auxiliary constituents. Sixteen different fault trees for the system were examined in the analysis, one for each "power state," with each state being defined as having power either available or unavailable at four 480V busses. Only eight power states were actually used because two of the four busses are tied together.

In the review, a simplified power dependency fault tree was constructed for each of the four busses to ascertain the subtle AC and DC interactions. For example, each bus can be powered by a diesel generator (two busses are powered by one) which then requires DC control power and a fuel oil pump, etc. These trees were then compared to the analysis presented in the IPPSS. No discrepancy was found. (It must be noted, however, that not all of the electrical system dependencies are modeled in the electrical system fault trees. Specifically, service water cooling of the diesel generators is omitted here. This results from the methodology employed in the IPPSS wherein the failure probability of the service water system (as well as other systems) is computed conditionally, that is based on the electric power state. This can be cumbersome and confusing initially but presents no great obstacle in understanding the analysis).

In addition to the system interactions, three additional, specific items were investigated. The first was the handling in the analysis of the combinations of signals which could be present with an initiating event. These signals are safety actuation, undervoltage, and reactor trip. Various combinations of these signals strip emergency loads from the busses which then must be reloaded. The IPPSS cites the emergency procedures which address this and includes as a failure, the failure of the operator to reload. Furthermore, the combinations of signals which cause the stripping appear to have been modeled properly.

The second specific issue examined in the review was that of multiple inverter failures. Lightning strikes at several plants have caused such failures, and the IP-2 electrical system was studied to ascertain if it could be a potential problem. To fail a single inverter in this fashion at least two separate circuit breakers would have to fail to open on the power surge. In addition, the diversity of instrumentation for ESF actuation and the redundancy of instrumentation among the four inverters indicate that simultaneous failure of at least six circuit breakers would be necessary to create a potential problem. Unable to find common cause data for such, we feel that this has probabilistically negligible impact on the risk from IP-2.

The third specific issue addressed in the review was that of common cause failures of the three diesel generators. An Oak Ridge study as part of TAP-A44 derived a "generic" probability for common cause diesel generator failure of 7(-4) for two diesel generators and 2(-4) for three. No comparable common cause values are found in the IPPSS.

With the application of the generic common cause values to the IPPSS unavailabilities, the system failure probabilities increase by about 50% or less:

Condition	IP-2 Valve	IP-2 Valve w/Common Cause
Failure of 2 DGs	1.4 (-3)	2.1 (-3)
Blackout	6.2 (-4)	8.2 (-4)

Thus, the onsite AC analysis appears to be reasonable.

In addition to the three diesel generators, the IP-2 emergency power system has three gas turbines available. Although we feel that the IPPSS gives too much credit to recovering failed diesel generators, the recovery analysis is relatively insensitive to such recovery because of the presence of the gas turbines.

2.4.1.2 IP-2 Reactor Protection System Fault Tree

The fault tree for the reactor protection system (RPS) of IP-2 was reviewed and found to be acceptable, except that the test and maintenance portion of the analysis appears to contain an error. However, its numerical significance is negligible.

The mean failure probability of the system is given as 2.01(-5), and this is comprised mainly of three contributors: random hardware failures of two trains of the trip system with a probability of 9.6(-6), failures in test and maintenance with a probability of 6.2(-6), and failure of the rod control cluster assemblies to enter the core with a probability of 3.8(-6). It must be noted that the analysis presented does not address manual scram. This is important because, although the first two failure mechanisms above are recoverable by the operator pushing the scram button, the third one is not, and it represents 19% of the total.

The probability for the rod control clusters failing to enter the core is derived from industry data and appears to be conservative. The event is failing to fully insert and no attempt is made to analyze the difference between, e.g., 90% insertion and 0% insertion. Both are considered to be system failures.

The random hardware failures in two trains consider shorting to power and ground, reactor trip or trip bypass breakers failing closed, and relay and logic matrix failures. This particular portion of the analysis follows that of WASH-1400.

The test and maintenance part of the analysis uses actual plant experience to determine the mean outage times for a train, and then this data was "ANDed" with random hardware faults in the other train. Finally, the value was doubled to account for either train undergoing the T&M. The problem in the analysis arises with the description used for the random hardware faults of one train with the other in T&M. In this case, only the probability of shorting to ground was considered although other failure mechanisms are possible. The logic matrix and relay failures contribute the same here as they do for the above two train case. However, for the T&M situation, both the reactor trip and bypass breakers would have to fail closed concurrently to cause the operating train to fail its function (see IPPSS Figure 1.5.2.2.2.-3) whereas above, the failure of either would cause train failure.

It is fortunate for IPPSS that these ignored failure modes have negligible (< 1%) contribution to RPS failure. Nevertheless, the description in Section 1.5.2.2.2.4.3 is wrong.

The common cause portion of the analysis uses the probability of a single instrument channel failing as the common cause miscalibration probability of a set of instruments. This approach may or may not be conservative because the calibration procedures do not appear to have been thoroughly analyzed. It must be noted, however, that the RPS has considerable diversity and redundancy in instrumentation and further that, should the common cause probability used be low by as much as an order of magnitude, the overall RPS failure probability would increase by less than 25%.

2.4.1.3 IP-2 Safeguards Actuation System Fault Tree

The analysis of the IP-2 safeguards actuation system is separated into two parts: safety injection (SI) and containment spray (CS) actuations. The analyses of both contain assumptions which could affect the unavailability values presented in the IPPSS.

Manual actuation is explicitly excluded from the fault tree analysis (but is considered at the event tree level). For other plants, this exclusion at the fault tree level could pose problems because different size LOCAs would result in different sets of actuation instrumentation initiating the system (e.g., for many plants, high building pressure would not actuate high pressure injection for small LOCAs as soon as needed). At IP-2, however, the same set of parameters, in general, cause actuation for all LOCAs. This is due to the relatively low trip level (2 psig) for the high building pressure sensors. In response to NRC questions on the IP-2 FSAR, the plant demonstrated that the building sensors would actuate before the presurrizer would empty for various LOCA sizes.

Test outage time is analyzed conservatively in that the channel undergoing monthly testing is assumed unavailable for the entire test duration (1 to 6 hours). The channel, however, is not unavailable for this entire period, and, even if it was, the tester could quickly switch it to the operating mode. (It should be noted that cut sets with a channel unavailable due to testing dominated the system failure probability).

The common cause value described in Section 1.5.2.2.3.4 is based on the value of a single instrument channel failing which is then used as the probability of common cause miscalibration of a set of instruments. This value may or may not represent such a failure possibility because the calibration procedures do not appear to have been closely examined.

The possibility of failure to restore the channels after testing is not adequately addressed in Section 1.5.2.2.3.4.6. While restoration following the monthly tests is discussed (and the analysis is correct), restoration following refueling outage tests is not analyzed. That is, no mention is made of the common cause human error that occurred at the Hatch reactor wherein the building pressure sensors were not restored following the dead-weight testing performed during the refueling outage. In discussions with the IPPSS analysts, we discovered that they had, in fact, considered such a failure to restore for the sensors. They had determined that this was of negligible probability at the IP plants because of the procedures, and we concurred with their assessment.

The two subsystems (SI & CS) share some common instrumentation but are analyzed separately. Because of the diversity and redundancy of the overall instrumentation for each, the probability of common failure due to this is negligible, and, hence, this part of the analysis is acceptable.

In conclusion, miscalibration should be better analyzed. With such, however, the overall failure probability might not change because of the testing conservatism contained in the present analysis.

2.4.1.4 IP-2 High Pressure Injection System Fault Tree

The IP-2 high pressure injection (HPI) system fault tree was reviewed, and problems were encountered with the IPPSS analysis. This is quite important because in one operating mode or another HPI is part of every IPPSS event tree except ET1, the tree for a large LOCA initiating event. Following are comments regarding the system in general after which the medium and small LOCA success criteria cases will be examined.

The system is analyzed by segmenting it into "supercomponents". Supercomponent A (see IPPSS Figure 1.5.2.3.1.-3) is the HP suction line from the RWST and consists of three valves (see Figure 1.5.2.3.1-4): manual valve 846 which is locked open, motor-operated valve 1810 which is deenergized open, and check valve 847. The first of these, 846, should not be a part of this tree, given the structure of the IPPSS event trees, because it is not unique to this system. This valve is also in the suction line to the LP pumps and is thus common to both systems. Therefore, for the situation where both the HP and LP injection systems are required, the handling of this valve in the IPPSS analysis results in accounting twice for the failure probability of valve 846. (Its failure probability is given as 2.64(-5) on p. 1.5-479.) The only event tree thus affected is that of the medium LOCA, ET2.

At the given failure probability for valve 846, the effects of this error are slight because, as shown below, it is but a small contributor to the overall system failure probability. It must be noted that although valve 846 is not alarmed, it is open with its handwheel removed and is on the startup and monthly check-off lists. In addition, in supercomponent A, MOV 1810 not only is deenergized open, but its position is tested monthly as part of the HP pump test. As with all MOVs in engineered safety systems, its position indication is given in the control room, and is alarmed should its position be "off normal".

Of more significance is the modeling of supercomponents B, C, and D which contain, respectively, pumps 21, 23, and 22, which is the swing pump and can inject water into either injection path 16 or 56. The analysis of supercomponents B and C appears to be correct but that of D does not. As stated in the IPPSS, MOV 851A, which connects pump 22 discharge to header line 56, will close if pump 21 fails. Similarly, MOV 851B, which connects pump 22 discharge to header line 16, will close if pump 23 fails. This action of the two valves is not explicitly modeled in the fault tree. It appears to be ignored in the medium LOCA case and "conservatively" analyzed for the small LOCA case. That is, the analysis actually considers a 1-out-of-2 pump case with the failure of pumps 21 and 23 failing the system because of the actions of the two valves. Unfortunately, the latter analysis is not conservative because additional failure mechanisms were not considered. It must be noted, however, that the operator action of opening either, or both, of these valves is also not modeled. In fact, no operator actions are modeled for this system, including errors.

On p. 1.5-483 of the IPPSS, it states that "... the procedures of the monthly and quarterly tests appear to minimize human error...." We feel that this is not the case because the pump tests are not staggered and are, in practice, performed by the same people on the same day. Thus, we feel that there is a strong human error dependency possible for this system.

HPI for Medium LOCAs

Event tree ET2 requires HPI with success criteria of two of three pumps injecting into two of four headers. (Header lines 16 and 56 mentioned above split into two headers each). IPPSS calculates an unavailability for this system mode as 4.1(-4). The analysis allows for dependence among the pump trains by adopting a subjective β-factor of 0.014. Literature was searched by us to determine the applicability of this value. Data presented in Atwood (EGG-EA-5288) indicate that, for failures and command faults for ESF standby pumps, a common cause probability of 5.2(-4) should be used for a system which requires two of three pumps to operate and has monthly testing. With the pump train failure probability given in the IPPSS as 7.02 (-3), this results in a β-factor of 0.074, more than a factor of five greater than that used in IPPSS. With the substitution of the common cause probability taken from Atwood for that given in the IPPSS (and correction of a few numerical errors), the system unavailability is recomputed to be 6.9(-4).

This probability may be non-conservative for three reasons. First, the dependency parameters taken from the referenced report by Atwood are medians, not means, and hence the data presented biases the results obtained to the low side. Secondly, the data given in Atwood are taken from the whole nuclear industry, not just from IP-2. Hence, for example, the effects of staggered and non-staggered tests are included while IP-2 does not stagger tests, as noted above, and might therefore have stronger dependencies among the pump trains. Thirdly, the use of a β-factor tends to make the error in modeling supercomponent D (the automatic closure of valves) less significant, but it does not eliminate the error. However, because IPPSS did not consider operator actions to recover the problem, this review did not examine the problem further.

HPI for Small LOCAs

Event tree ET3 requires HPI with success criteria of one of three pumps injecting into one of four headers. In addition, this model is part of every other event tree analyzed except for ET1 and ET2. For event trees ET4, ET5, and ET6, the model is a part of event SA2, safety actuation and high-head injection. For event trees ET7 through ET12, it is part of event OP-2, operator establishing feed and bleed. For the ATWS event tree, ET13, it is part of event OP-5, manual actuation of feed and bleed with emergency boration.

IPPSS attempted to conservatively analyze this HPI success criteria case. The analysis considered a one of two pump situation and therefore strove to ignore the problem with the modeling of supercomponent D. However, here IPPSS used no β-factor, i.e., they assumed that there was no dependency between the two pump trains whereas for the medium LOCA model there was dependency! The failure probability for the one of two case is given as 1.86 (-4).

This review used data from the Atwood reference to determine that, generically for this type of system, a dependency among all three pump trains exists such that the failure probability for all three trains is 3.6(-4), which yields a β-factor of 0.051 for this system alignment. (It must be noted that the β-factor method assumes that the third pump will fail if the second fails.) As stated above and for the same reasons, this probability is probably nonconservative. However, with the use of the common cause probability, and the changing of the one of two situation to a one of three situation on p. 1.5-486, the system failure probability becomes 5.0(-4).

2.4.1.5 IP-2 Low Pressure Injection System Fault Tree

The IP-2 low pressure injection (LPI) system fault tree was reviewed, and the only major difficulty encountered with the model was that of the common cause value used in the analysis. Of minor import, manual valve 846 is considered an unique part of the LPIS, but as discussed in Section 2.4.1.4 of this report, in reality it is not. However, the only event tree this error affects is ET2, and the probability of both the LPIS and HPIS failing is overestimated by 2.6(-5) in the IPPSS. The LPIS fault tree is also used in event tree ET1, which is initiated by a large LOCA.

As for the HPIS, the LPIS is divided into supercomponents for the purpose of analysis (in fact, this method is used in the IPPSS for all fluid systems). The analysis considered common cause failures between supercomponents B and C, containing RHR pumps 21 and 22 respectively, and supercomponents E and F, containing motor-operated valves and heat exchangers 21 and 22 respectively. As for the HP analysis, a β-factor of 0.014 was subjectively assumed. Supercomponent B(C) was computed to have a failure probability of 6.5(-3), and E(F), a failure probability of 2.3(-3). Hence, with the use of the 0.014 β-factor, the probability of failure for the Boolean combination of (B AND C) OR (E AND F), which becomes (βB OR βE), was calculated to be 1.23(-4). Overall, the LPIS has a failure probability of 8.7(-4). (It should be noted that motor-operated valve 882, which is deenergized open and is a suction valve common to both pumps, contributes 4.9(-4) of the total. MOV 882 is verified open only at refueling outages. If IP-2 altered its testing, the fault exposure time of this valve could undoubtedly be significantly reduced.)

As for the HPIS review, the pump common cause data presented compiled by Atwood (EGG-EA-5288) were used to determine the applicability of the 0.014 value. In fact, the failure and command fault probability for two pumps in an ESF standby system requiring one of two to operate for success and having monthly tests was found to be 5.7(-4). This suggests a β-factor of 0.088 which is more than six times greater than that used in the IPPSS. As stated above in the HPIS section, even with the application of this β-factor in the model, the results may be non-conservative because the data presented in the reference are medians, not means, and IP-2 does not stagger the pump tests.

To partially account for the non-conservatism, the 0.088 β-factor was applied to the E and F supercomponents as well as those of B and C. Hence, the failure probability of the LPIS is re-estimated to be 1.2(-3).

2.4.1.6 IP-2 Accumulator System Fault Tree

With one exception, the fault tree constructed for the IP-2 accumulator system is correct. The exception is that of the four check valves located most downstream from the accumulator. Figures 1.5.2.3.3-3 and 1.5.2.3.2-3 in the IPPSS show the configuration for the four accumulator/cold leg piping connections. For example, check valve 897A is analyzed as being an independent part of the accumulator system and also an independent part of the low pressure injection system whereas its plugging is a common mode failure of both systems failing to inject water into cold leg 1. (The arrangement for the other three cold legs is identical).

The mean for one of three check valves plugging is given as 2.1(-4) with the mean system unavailabilities being 1.9(-3) for the accumulators.

Because the success criteria for large LOCA emergency core cooling injection requires both the accumulators and low pressure injection, these two systems are ORed together for function failure. Hence, the result of not considering these valves as part of both systems is to increase the combined failure probability by about 5% more than it should be. (The present fault tree models account twice for the failures of these check valves).

2.4.1.7 IP-2 Recirculation System Fault Trees

The hardware portions of the recirculation systems analysis were herein reviewed. The human error contributions, which in the IPPSS are the dominant causes of failure, are examined in Section 2.5 of this report. In the IPPSS, there are four types of recirculation considered: high pressure, low pressure, containment spray, and hot leg low pressure.

<u>High Pressure Recirculation (HPR)</u>

The mean failure probability for this system is calculated as 6.8(-4) in the IPPSS of which 3.9(-4) results from human error and 2.9(-4) results from hardware failure. For HPR, either the recirculation or RHR portion of the LPRS must be operating as well as component cooling water. These dependencies are explicitly modeled on the fault tree. In fact, the presented fault tree is quite comprehensive. However, the computation of system unavailability does not use the presented fault tree. Rather, conditional probabilities were calculated for the system based on whether or not the containment fan system (hence, service water which cools the fan coils and component cooling water) is working. This analysis is currently being modified by the IPPSS analysts. However, the material presented below probably will not need amending with the modification.

The analysis presented in the IPPSS explicitly models the common cause failure possibility among the HP pump trains but, inconsistently, ignores it for the recirculation and RHR pump trains. As will be shown below, we believe a better estimate for the LP portion of recirculation (either through the RHR or recirculation pumps) is 4.4(-4), and this includes common cause failures. Thus, we need here to only re-estimate the HP portion of the system to yield a failure probability for HPR.

The pump train failure data given in the IPPSS for the HP system are 6.4(-3) to start and 3.8(-4) to run, given start (1.6(-5)/hr times 24 hrs.). The Atwood reference was searched for data concerning common cause failure of three standby pumps to run and two to start (i.e., HPI with one pump is assumed to have succeeded). The recomputed failure probability of the pumps is 3.5(-4). In addition to the pumps, either MOV 888A or 888B can fail (they are valves in redundant paths from the RHR heat exchangers to the HP pump suction, see IPPSS Figure 1.5.2.3.4-6). The datum given for a MOV failing to open is 2.3(-4). If the β-factor of 0.1, the same as for the valves in the LPR analysis below, is used here, the total HPRS hardware failure probability for the HPRS becomes 8.2(-4).

Low Pressure Recirculation (LPR)

The failure probability for LPR at IP-2 is given as 5.5(-3) of which 5.3(-3) is due to human error. Table 1.5.2.3.4-18 of the IPPSS gives the dominant failure modes of the system. Here the recirculation pump trains have a β-factor assigned of 0.014 whereas for HPR above, they had none. Also the table is noteworthy for the absence of the RHR pumps failing. Rather, the constructed model assumes, and the results in the table show, that should the recirculation pumps fail, the only failure in establishing RHR flow is the failure of the operator to initiate the switchgear from the recirculation to the RHR pumps. A probability of 0.26 is assigned to this, but the fact that the RHR pumps themselves could subsequently fail is neglected. It must be noted that the RHR failure probability is significantly less than 0.26, but the description should have stated why it was being ignored.

To re-estimate the probability of this system failing, β-factors of 0.088 and 0.1 were used for the pump trains (both recirculation and RHR) and valve sets, respectively. The former is that used for the LPIS review (see Section 2.4.1.5), and the latter is a subjective estimate on our part. Then, the failure probability for the LPRS can be expressed as:

$$Q=(\beta_1 Q_{\substack{recirc\\pumps}} + \beta_2 Q_{\substack{MOV\\1802A,B}})\ (\beta_1 Q_{\substack{RHR\\pumps}} + 2Q_{\substack{MOV\\885A,B}} + Q_{\substack{MOV\\1805}} + 0.26)$$

$$+ \beta_2 Q_{\substack{MOV\\822A,B}}$$

where β_1 is 0.088, β_2 is 0.1, and the failure probabilities for the others are taken directly from the IPPSS analysis (except the probability of the 885A,B valves failing are multiplied by 0.1 to allow for recovery by the operator because they are outside containment). This results in a hardware unavailability for LPR of 4.4(-4). (It must be noted that the above equation assumes that $Q_i^2 << \beta_j Q_i$, which in this case, it is.)

Containment Spray Recirculation (CSR)

The failure probability of the CSRS is estimated in the IPPSS as 1.5(-3) of which 99% is human error. The only other failure possible, given that LPR is working, is for two motor-operated valves (889A,B, see Figure 1.5.2.3.4-6) to fail closed. IPPSS evaluates this with its β-factor of 0.014 which results in a hardware contribution of 3.7(-5). If the 0.1 β-factor is used, this hardware contribution increases to 2.3(-4).

Hot-Leg Recirculation

The hot-leg recirculation considered here is not that of normal shutdown cooling. Rather, it is initiated approximately 24 hours after the initiation of the accident, and thus its use presupposes the success of LPR. Because LPR can be maintained, it is felt that this portion of the analysis is not risk-significant and was hence not reviewed in detail. However, it appears that the analysis is acceptable.

2.4.1.8 IP-2 Containment Spray Injection System Fault Tree

The fault tree for the IP-2 containment spray injection system is inconsistent with the analyses of the other systems. Section 1.5.2.3.5.4.1.3 of the IPPSS states: "Common cause failures of the same type of component in different trains could occur, but the probability in a standby system that is tested monthly and can be maintained during reactor operation is judged to be very small.... Therefore, no common cause contribution has been assigned to this system." The standby condition and the test and maintenance characteristics of the CSI pumps are no different than those of the HPI and LPI pumps. Thus, the cited IPPSS statement is inconsistent with other IPPSS analyses.

Other than the common cause discrepancy, the CSI systems analysis appears to be correct. Hence, the rest of this section shall examine the effect on the system unavailability of including pump common cause failure considerations.

IPPSS gives the CSIS unavailability as 7.5(-5) (sic) with random hardware failures contributing 5.1(-5), maintenance on one train with failures in the other contributing 1.1(-5), and operator error is not restoring from a test condition contributing 1.4(-5). (Unavailability due to testing itself is several orders of magnitude lower.) The derivation of the latter two values appears to be correct, but the neglecting of the pump common cause failure possibility makes the first value suspect. The failures of a spray pump to start and to run for two hours, given a start, is given as 6.5(-3). For a pump train, this value increases to 6.9(-3), with a variance of 7.4(-6). Thus, the probability of random hardware failures of both trains is 5.5(-5) (not the 5.1(-5) given in the IPPSS).

As presented in Section 2.4.1.5 of this report, for two pumps in standby, there is a common cause fault probability of 5.7(-4), which yields a β-factor of 0.088. With the use of this common cause probability, the system failure probability becomes 6.0(-4), a factor of eight higher than that reported in the IPPSS.

2.4.1.9 IP-2 Containment Fan Cooling System Fault Tree

The fault tree constructed for the IP-2 containment fan cooling system is correct, given the assumption used in the analysis. Three assumptions could alter the calculated system unavailability, two of which would decrease the value and one of which, increase it. Of the former type, the success criterion assumed in the analysis is that of the IP-2 FSAR, that three of the five fan cooling units must operate to achieve system success. Other PRAs (such as ANO-1) have found that the success criteria for fan systems which are reported in safety analysis reports can be conservative, instead of realistic (see Section 2.3). Hence, the calculated fan system failure probability may be conservative. The second conservative assumption is that the analysis does not give credit for manual actuation of the system; only automatic actuation is considered. Because of the relatively long time available for operator recovery actions to restore system function and prevent containment overpressurization, manual actuation is viable. Failure of automatic actuation, however, is a small contributor to the overall system unavailability.

The assumption which is potentially non-conservative is that the charcoal filter beds will not plug with airborne debris during the course of the accident. This assumption has been made in other PRAs (again, such as ANO-1) but has been a subject of sensitivity studies in them because the phenomenology is not currently well-defined. (NRC programs, e.g., ASEP, are looking at this to see if it is risk significant.) The sensitivity of the overall risk to this assumption was not done in IPPSS, but is investigated in Section 4.2 of this report.

2.4.1.10 IP-2 Component Cooling Water System Fault Tree

The component cooling water (CCW) system at IP-2 is capable of cooling any heat source by water discharged from any pump. That is, the system is totally headered together. Also, during normal operation, two of the three pumps are running.

The IPPSS gives the failure probability for CCWS as 1.0(-5) for the power condition of all busses available, 6.1(-4) for the condition of 1 bus lost, and 6.5(-3) for the condition of power at two busses lost. The first case is for the situation in which the CCW pumps, which were operating prior to the initiating event, do not trip because of the initiating event. The last two cases are for the situation in which the CCW pumps are tripped as a result of the initiating event. Either loss of offsite power or a safety injection signal will cause the pumps to be tripped and then sequentially loaded.

There are two principal areas of disagreement we have with the presented analysis. First, the effects of a pipe break on the system performance are not analyzed. As demonstrated in Section 4.6 of this report, this is non-conservative. The second area of disagreement is that common cause effects are assumed to be negligible, which makes the CCWS analysis inconsistent with the rest of the IPPSS. Generic common cause data (Atwood, EGG-EA-5288) were again examined. For pumps such as those in the CCW, the data suggest a common cause failure probability of two of two failing to start and then failing to run for 24 hours of 2.9(-5). (Failure to run alone is 2.4(-5)). In addition, for the failure to start and then run for 24 hours, the data suggest 1.44(-5) for the common cause failure probability for a one of three pump system. Both values are needed to review the CCWS because, without pump trip, two CCW pumps will continue to operate and, with pump trip, one of three is required to start and operate. In the former situation, should both pumps fail, then the third is required to start. IPPSS gives the failure probability for this third pump as 6.54(-3).

The equation presented on p. 1.5-786 of the IPPSS can be used to evaluate the pump failure contribution to system failure for the various situations. (For all power conditions and initiating events, the remainder of the CCWS contributes 5.7(-6) to the overall failure probability). With the use of the above information and the IPPSS equation, the overall system unavailability, for the case of no pump trip and power available at all busses, is 8.3(-6), not the 1.0(-5) presented in the IPPSS. (It appears that failure to start data was used instead of failure to run.) For the situation with pump trip and power available at all busses, the system unavailability is re-estimated to be 2.0(-5).

For the situation where the pumps trip and power is lost to one bus, the equation on p. 1.5-789 can be used to determine pump train contribution to system unavailability. With the use of the above common cause datum, the CCWS failure probability for this case is estimated to be 2.9(-5) which is significantly less than the 6.1(-4) reported in the IPPSS. For the situation of pump trip and power available to only one pump (loss of power to pumps 22 and 23 is the worst case because they are powered by the same diesel generator), the pump failing to start or to run is 6.54(-3) which is the value reported by the IPPSS. However, for this bounding condition, they did not add in the unavailability of pump 21 due to maintenance which is 1.39(-3) as reported on p. 1.5-776. Therefore, for this last situation, the system unavailability should be 7.93(-3).

2.4.1.11 IP-2 Service Water System Fault Tree

The service water (SW) system consists of two subsystems, each having three pumps: the nuclear header and the conventional header. Each subsystem is completely headered together so the analysis complications are lessened. The analyses of both were reviewed and found suspect in certain respects. Success of the system is defined as two nuclear header pumps operating and one conventional header pump. (In the analysis of the loss of offsite power initiating event, IPPSS committed two mistakes. First, they kept to this success criterion although all three diesel generators can be cooled by one nuclear header pump, and secondly, they added the two subsystem unavailabilities together whereas the diesel generators do not require cooling at all from the conventional header.)

IPPSS gives the unavailabilities for the two subsystems as

Power Condition	Nuclear Header w/Safety Actuation	Nuclear Header w/o Safety Actuation*	Conventional Header
All Power Available	2.4(-4)	4.6(-5)	5.4(-6)
Power Lost at 1 bus	1.8(-2)	7.8(-5)	5.9(-4)
Power lost at 2 busses	1.0	7.0(-3)	7.9(-2)

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*with loss of offsite power

Two problems arise from the analysis. First, the three gates to the intake structure are assumed to fail, by plugging, completely independently of each other. (It should be noted that the SW pumps normally take suction from only one, but upon a safety actuation signal, the other two have doors which are to open for the SW pump suction.) Secondly, common cause failures among the pumps of each of the two subsystems are assumed to be negligible.

As to the former concern, IPPSS uses a mean probability of the intake screen plugging of 2.66(-5) per hour and then combines this with the failure of either of the two doors to open or plugging of the screens of the other two intakes. All of these failures are assumed to be independent of each other, and the intake structure unavailability is estimated to be 6.0(-10) over the 24 hour period of the accident.

The visit to IP-2 revealed that the three screens are side-by-side, each about 20 feet in width. They are cleaned daily, in succession. Because of their proximity and the sequential cleaning routine, it is felt that a strong dependency exists among the three screens. NSIC data were reviewed to ascertain if nuclear plants are susceptible to plugging of the service water system. Six possibilities were found at Duane Arnold, Hatch, ANO-1 (twice), ANO-2, and San Onofre 1. None of these resulted in complete SWS failure prior to operator-initiated safe shutdown, but the instances do indicate the possibility of a common cause SW failure in the integrity of its source.

minutes unless the operate trips it. The IPPSS analysis handles all three pumps as being identical in this respect, that, if the condensate storage tank water supply be lost, the operator has 30 minutes to align in the city water supply. No other human actions are considered given CST supply failure.

Secondly, the turbine-driven pump only cranks to a minimum speed automatically. The operator must manually bring its speed above the minimum. This is not modeled.

Thirdly, common cause failures between the two motor-driven pumps are ignored. As shown in previous sections, this tends to be non-conservative. (The β-factor method presented by Atwood (EGG-EA-5288) cautions against using the method among dissimilar components, e.g., motor- and turbine-driven pumps.)

To evaluate the effect of these errors, a simplified system fault tree was constructed using the supercomponents identified in the IPPSS and, for the most part, the data presented there. It was assumed that, if the CST source failed, the turbine-driven pump would automatically fail with no recovery potential. Furthermore, in consistency with the IPPSS human error probabilities for the AFWS, a probability of 0.07 was for that the operator would not increasing the speed of the turbine-driven pump. Finally, from the Atwood reference given above, it was determined that the β-factor for the two motor-driven pumps is 0.204. With the use of these values, the system unavailability was re-estimated to be 3(-5) for the all power available case and 2.3(-2) for the blackout case.

2.4.2 Fault Trees of Indian Point 3

Section 1.6 of the IPPSS presents the systems analysis for the Indian Point 3 reactor. The review of these analyses is presented below. Unless otherwise noted, system failure probabilities cited herein are for the case of all power available, which is of primary concern except for a few systems. By the nature of the review, only those areas of disagreement are discussed.

2.4.2.1 IP-3 Emergency Electric Power System Fault Tree

The IP-3 emergency electric power system is very similar to that of IP-2. The principal difference is that at IP-3, there is no automatic transfer to a backup DC supply for the diesel generator starting requirements. The review of the IP-3 system was identical to that of IP-2 (see Section 2.4.1.1), and the IPPSS analysis appears to be reasonable. For example for the blackout case, IPPSS reports a failure probability of 1.0(-3) whereas this becomes 1.2(-3) with the inclusion of the generic common cause diesel generator failures.

2.4.2.2 IP-3 Reactor Protection System Fault Tree

The fault tree for the reactor protection system (RPS) of IP-3 was reviewed and found to be acceptable, with the same reservations as those expressed in Section 2.4.1.2 of this report.

The mean system failure probability is found to be 3.93(-5) with the difference between the IP-3 and IP-2 values resulting from the different operational histories of the two plants. For example, the mean test unavailability at IP-3 is given as 8.54(-3) while it is 5.97(-3) at IP-2. Similarly, IP-3 has had far fewer demands of its RPS than has IP-2. (This results not only from the longer operating time of IP-2 but also from the greater number of transients which IP-2 has experienced). As given in the IPPSS, IP-3 has had 0 failures of rod cluster assemblies to fully insert in 1908 demands whereas IP-2 has had 0 failures in 6784 demands. Thus, IPPSS gives the mean probability of this failure at IP-3 as 9.2(-6), which means that 24% of the RPS failures are not recoverable by pushing the manual scram button.

2.4.2.3 IP-3 Safeguards Actuation System Fault Tree

The comments for the IP-2 Safeguards Actuation System Fault Tree, Section 2.4.1.3 of this report, are applicable here as well.

2.4.2.4 IP-3 High Pressure Injection System Fault Tree

The same reservations we expressed in Section 2.4.1.4 about the IP-2 HPI analysis also hold here for the IP-3 HPI system. The analyses presented in IPPSS Section 1.5.2.3.1 for the IP-2 HPIS are identical to those presented in Section 1.6.2.3.1 for the IP-3 HPIS with the exception of the plant specific data.

The failure probabilities presented for the HPIS are 1.8(-4) for the medium LOCA success criteria and 1.3(-4) for the small LOCA success criteria. The latter value results almost exclusively from the failure of one of the three valves in the RWST suction line. The pump train failure probability is given as 1.5(-3) for the IP-3 HPIS whereas it was 7.0(-3) for the IP-2 HPIS.

Our review again reanalyzed the system failure probability for the two different success criteria, particularly with respect to pump train dependencies. With the use of the Atwood data (EGG-EA-5288), the recalculated failure probabilities become 6.8(-4) for the medium LOCA model and 4.9(-4) for the small LOCA model.

2.4.2.5 IP-3 Low Pressure Injection System Fault Tree

The review of the IP-3 low pressure injection (LPI) system fault tree showed it to be the same as that for IP-2 (see Section 2.4.1.5) except for the handling of common cause failures and the different data used. The failure probability of the LPIS for IP-3 is given as 8.1(-4) in the IPPSS.

As to common cause, the presented analysis simply ignores it which seems to indicate that its omission is an oversight. With the data presented in the IPPSS and the use of the β-factor of 0.088 used by us for the IP-2 LPIS review, the IP-3 LPIS failure probability is re-estimated to be 9.3(-4).

(It must be noted that motor-operated valve 882, which in the common suction line for both LP pumps and is normally de-energized open, contributes 6(-4) to the system unavailability, or roughly two-thirds of the total. This results primarily because the valve is verified open only at refueling outages. A change in procedures could surely significantly reduce its fault exposure time.)

2.4.2.6 IP-3 Accumulator System Fault Tree

The fault tree constructed for the IP-3 accumulator system is correct, with one exception. The exception is the same as that noted in Section 2.4.1.6 and concerns the check valves shared by the accumulator and low pressure systems. The discussion presented in the referenced section applies here as well.

2.4.2.7 IP-3 Recirculation System Fault Trees

The same comments made in the review of the IP-2 recirculation system (Section 2.4.1.7 of this report) apply here as well. The differences which occur are the result of the different data used for IP-3 than for IP-2.

The failure probability for HPR is given as 4.1(-3) of which 3.9(-4) is operator error and 3.11(-3) is failure of the HP pumps to operate during the 24 hours duration. With the replacing of the IPPSS β-factor with information gleaned from Atwood and with the re-estimated LPR failure probability given below, we estimate the hardware contribution to HPR unavailability should be 8.4(-3). However, as is stated in Section 3.3.1 of this report, we believe this value is false because the IP-3 HP component failure probabilities are a result of overapplication of the data.

For the LPRS, if the IP-3 specific data is used and applied to the equation given in Section 2.4.1.7, the new estimation for the LPR hardware failure probability becomes 1.9(-4). Similarly, the CSR hardware failure probability becomes 1.5(-4).

2.4.2.8 IP-3 Containment Spray Injection System Fault Tree

The comments made for the IP-2 CSIS analysis apply here as well (see Section 2.4.1.8). The identical rationale for neglecting common cause failures is cited in IPPSS Section 1.6.2.3.5.4.1.3 as in Section 1.5.2.3.5.4.1.3.

The system unavailability is given as 3(-5) (sic) with random hardware failures of the two trains contributing 1.3(-5), operator error in restoring from test contributing 1.4(-5), and one train out for maintenance with hardware failures in the other contributing 4.5(-6). The human error probability is identical to that used for the IP-2 CSIS analysis, and the maintenance experience of the two plants is nearly so (IP-3 gives a maintenance unavailability of 7.3(-4) whereas at IP-2, the value is 8.1(-4)). The major difference in the two system failure probabilities is in the failure probabilities of pumps failing to start and failing to run for two hours, given start. Here, that probability is 1.4(-3) with a variance of 1.2(-6), and for IP-2 it was 6.5(-3) with a variance of 7.4(-6). Similarly, for a pump train, the IP-3 hardware failure probability is 3.1(-3) with a variance of 3.6(-6), and the IP-2 values were 6.9(-3) with a variance of 7.4(-6).

If the 5.7(-4) common cause failure probability is used (see Section 2.4.1.5), the IP-3 CSIS failure probability becomes 6(-4), a factor of twenty greater than that reported. Thus, it would seem that industry experience indicates that pump common cause failures are very important for this system.

2.4.2.9 IP-3 Containment Fan Cooling System Fault Tree

The fault tree constructed for the IP-3 containment fan cooling system is correct, given the assumptions used in the analysis. The assumptions are the same as that for the equivalent IP-2 system and are discussed in Section 2.4.1.9.

The difference between the calculated fan system unavailabilities for IP-2 and IP-3 is attributable to different component failure and maintenance histories, as well as differences in actuation, at the two plants. For example, as to the former, valve failure experience is different at the two plants, here specifically in the air-operated service water discharge valves used in emergency operation. At IP-2, one of this type of valve has failed to open on demand during the history of the plant whereas there have been no failures of this type of valve at IP-3. Hence, two different data were used in the analyses of the two plants.

As to the differences in actuation, the safety equipment loads at IP-2 are stripped from their busses upon a safety actuation system whereas they are not at IP-3. Thus, the system unavailability at IP-3 is also lower than that at IP-2 because, at the former, the fans do not need to restart.

2.4.2.10 IP-3 Component Cooling Water System Fault Tree

The component cooling water (CCW) system at IP-3 is different than that at IP-2 in that the three pumps do not trip off except on a loss of offsite power, and in that event they are each powered by a separate diesel generator. The system is like that of IP-2, however, in that it is totally headered together. (Figure 1.6.2.3.7-4 in the IPPSS is in error. It shows valves 766C and D normally closed when, in fact, they are normally open.)

IPPSS gives the failure probabilities for the system as presented below (common cause failure is assumed to be negligible):

Power Condition	w/o LOP	w/ LOP
All Power	1.1(-7)	3.0(-7)
Power at 2 busses	1.8(-6)	3.0(-5)
Power at 1 bus	8.3(-5)	1.5(-3)

For IP-3, the failure of a CCW pump to start is 1.44(-3), and the failure of the pump to run, given start, is 9(-4). Furthermore, the unavailability of a pump due to maintenance is given as 1.84(-2) on p. 1.6-753. With the use of these data, the common cause data presented in Section 2.4.1.10, the equation presented on p. 1.6-762 of the IPPSS and the passive valve failure data given, the CCWS failure probability for the above cases is re-estimated to be:

Power Condition	w/o LOP	w/ LOP
All Power	1.4(-6)	9.8(-5)
Power at 2 busses	2.9(-5)	4.2(-5)
Power at 1 bus	1.8(-2)	2.0(-2)

The differences in our calculations and those of the IPPSS are attributable to our inclusion of common cause effects and their omission to account for maintenance for the two cases of power available at one bus.

2.4.2.11 IP-3 Service Water Fault Tree

The service water (SW) system at IP-3 is quite similar to that of IP-2. A major difference in the two systems is that the SWS of IP-3 has three backup SW pumps with a separate intake structure on the discharge canal. Thus, the screen common cause event of IP-2 does not exist at IP-3. It must also be remembered that a safety actuation signal does not strip loads at IP-3.

IPPSS gives the failure probabilities for the SWS as

Power Condition	Nuclear Header w/o LOP	Nuclear Header w/ LOP, 1 Pump	Conventional Header
All Power Available	3.1(-5)	8.3(-5)	7.2(-5)
Power Lost at 1 bus	5.0(-3)	9.3(-5)	1.3(-4)
Power lost at 2 busses	1.0	3.3(-3)	1.8(-2)

As with the IP-2 SWS analysis, that for the IP-3 SWS neglects common cause pump failures. Furthermore, the analysis of the special case condition (one nuclear header pump success criterion) ignores completely pump maintenance outages, as does the analysis for the nuclear header without LOP and power lost at one bus.

The data presented in Section 2.4.1.10 for common cause pump train failure probabilities are used here as are the data from IPPSS of a SWS pump having an unavailability due to maintenance of 1.47(-2), a failure to start of 1.43(-3), and a failure to run for 24 hours of 1.77(-3). With these data, the failure probabilities of the SWS are re-estimated to be

Power Condition	Nuclear Header w/o LOP	Nuclear Header w/ LOP, 1 Pump	Conventional Header
All Power Available	3.8(-5)	9.7(-5)	8.6(-5)
Power Lost at 1 bus	2.0(-2)	9.7(-5)	1.3(-4)
Power lost at 2 busses	1.0	1.8(-2)	1.8(-2)

2.4.2.12 IP-3 Auxiliary Feedwater System Fault Tree

The fault tree for the IP-3 auxiliary feedwater (AFW) system was reviewed and found to be in error in two of the three instances of errors in the IP-2 AFWS fault tree (see Section 2.4.1.12). The exception is that the turbine-driven pump at IP-3 automatically goes to full speed. IPPSS calculates the system unavailability as 2(-5) for the all power available condition and 1.6(-2) for the blackout condition.

As with the IP-2 AFWS review, a simplified fault tree was constructed for the IP-3 AFWS. With the use of the IP-3 data, the common cause failure datum for the two motor-driven pumps given in Section 2.4.1.12, and the assumption that the turbine-driven pump will fail if the CST supply fails, the AFWS unavailability was found to be essentially unchanged for the power available condition and to increase to 1.9(-2) for the blackout condition.

2.5 Human Reliability Analysis

2.5.1 Scope of the HRA Review

The human reliability analysis (HRA) portions of the Indian Point Probabilistic Safety Study (IPPSS) were reviewed and evaluated by the same Sandia human reliability analyst who reviewed the HRA portions of the Zion Probabilistic Safety Study (ZPSS). In this analyst's opinion, despite some shortcomings the IPPSS HRA represents a detailed, thoughtful, and objective attempt to analyze the most difficult-to-analyze system component--the human.

This HRA section is relatively short for three reasons: (1) The HRAs for both units at Indian Point are identical and are almost identical with the HRA made for the Zion plant, so much repetition seems unnecessary; (2) only four error terms from Section 3.0 have a major impact on the systems analysis in the Sandia evaluation of the IPPSS; and (3) the documentation behind the estimates of human performance used in the IPPSS is too sketchy to permit this analyst to properly evaluate many of the human error terms used in the PRA.

The next section (2.5.2) repeats the 11 areas of agreement/disagreement with the HRA in the ZPSS since they are applicable also to the IPPSS. In Section 2.5.3 each of these areas is discussed, with mention of any differences between the ZPSS and the IPPSS. Section 2.5.4 provides the only quantitative evaluation made here because there are only four error terms that have a significant impact in the Sandia systems analysis. Section 2.5.5 provides a short summary of comments on the IPPSS HRA. Finally, Appendix A lists some specific reservations about certain estimates or statements in the IPPSS, with emphasis on questions that cannot be answered because of lack of information in the IPPSS.

2.5.2 Areas of Agreement/Disagreement with the IPPSS

The following is a list of 11 areas taken from our review of the ZPSS which also apply to the IPPSS. The next section provides a description of each.

1) Incomplete and incorrect documentation of the HRA.

2) Use of large uncertainty bounds in the HRA.

3) Use of undue optimism in assessment of credit for human redundancy.

4) Use of optimistic assessments of human performance under stress, especially for cases of multiple problems.

5) Use of persons to estimate operator performance in place of simple measurements.

6) Lack of documentation on how expert opinion was used.

7) Incomplete documentation of data sources used for estimated human performance.

8) Use of optimistic assessments of dependence among tasks done by same person.

9) Possible insufficient consideration of common-cause failures from human errors.

10) Possible insufficient consideration of errors in restoring safety components after test, maintenance, or calibration.

11) Frequent use of conservatism in the HRA.

2.5.3 Description and Qualitative Assessment of the Areas of Agreement/Disagreement

This section discusses each of the 11 areas identified in the previous section.

1) Incomplete and incorrect documentation of the HRA.

As near as this analyst can determine, the HRA portions and estimates of human error probabilities (HEPs) and assumptions about human behavior and interperson interaction are identical for the two Indian Point units (2 and 3), and nearly identical with the HRA done for the ZPSS. In view of the generic data for HRA available to analysts, the near identity of the HRAs for all three plants should not be construed as a criticism. Apparently, the same personnel performed all three HRAs, and made the judgment that there was a very high degree of similarity in the operator behaviors required in these different PWRs for the task evaluated. Therefore, the basic HEPs and many assumptions about operating teams made in the ZPSS were applied without change to the same tasks in the IPPSS. For some other tasks, changes were made between the ZPSS and IPPSS, e.g., the giving of less credit for the STA (shift technical adviser) in the Indian Point plants to catch operator errors than in the Zion PWR because in the former plants the STA is not an SRO (senior reactor operator) as is the case in the latter plant. Such extrapolation can be warranted; this analyst is unable to evaluate this type of generalization of results because of the time limitation placed on this review.

While this analyst does not criticize the above generalization, the IPPSS should have made this procedure clear. In some cases, a reader might be led to believe that separate analyses were made in the Zion and Indian Point HRAs, when this does not appear to be the case. For example, on page 1.5-902, Section 1.5.2.3.9.4.4. "Human

Inaction," with regard to the Indian Point 2 Auxiliary Feedwater System, it is stated, "The probability of human inaction has been quantified into histograms based on discussions with operators, supervisory personnel, engineers, and after a review of the operating histories at other plants. The judgments take into account the high stress conditions in the control room during emergencies and the competing demands during the 30 minutes the operator has to perform his task." It may not be clear to the reader that the phrase "at other plants" applies to all of the foregoing--not just to a review of operating histories at other plants. The histograms on pages 1.5-903 and 904 are identical to those in the comparable section in the Zion PRA. It is reasonable to conclude that the PRA team decided that the Zion results could be applied to the Indian Point 2 PRA without modification. The same analysis was also applied to Indian Point 3.

The use of NUREG/CR-1278* in this PRA (as well as for the Zion PRA) for many of the estimated human error probabilities (HEPs) made it easy to find sources of such estimates. However, it was not possible to fully understand and evaluate the HRA by reading only those sections clearly labeled as "human reliability," "human error," or "human factors." Because of the lack of documentation and the difficult-to-follow format, it was frequently difficult to impossible to evaluate estimates of some HEPs and to track the translation of these HEPs into questions which combined both equipment failure and human error terms. In this respect, the Indian Point HRA is more difficult to track than the Zion HRA. Because of the time limitation placed on this review, the Sandia HRA specialist had to base many of his evaluations on the assumption that the operator tasks and equipment support and procedures at these plants are equivalent (highly similar) to those at the Zion plant with which he is familiar.

One major conclusion, then, is that HRA parts of a PRA should be documented in some systematic and reproducible manner, as is suggested in NUREG/CR-2254 **and implemented in the Arkansas Nuclear

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*Swain, A. D., and Huttmann, H. D., Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications (draft for interim use and comment), U.S. Nuclear Regulatory Commission, Washington D.C., Oct. 1980.

**Bell, B. J., and Swain, A. D., A Procedure for Performing a Human Reliability Analysis of Nuclear Power Plants, NUREG/CR-2254 (draft for interim use and comment), U.S. Nuclear Regulatory Commission, Washington, D.C., 1981.

One Unit 1 PRA.* Unless this is done, independent evaluation of the HRA portions of a PRA by others will be difficult to impossible.

2) Use of large uncertainty bounds in the HRA.

In general the IPPSS HRA makes use of wider uncertainty bounds than are found in NUREG/CR-1278. When their estimates of median HEPs are valid, the IPPSS HRA is more conservative, i.e., less likely to be optimistic about human interaction and intervention in a plant, than would be the case if they used narrower uncertainty bounds.

However, as in the ZPSS, it is stated in Section 0.15 (Vol. 1, page 0-99) that "We determine the log normal distribution by using the best estimate as the median and the upper bound as the 90th percentile, rather than the 95th percentile that the handbook recommends." ("Handbook" refers to NUREG/CR-1278.) Nevertheless, they forgot this conservatism and used the 95th percentile throughout the report. In the opinion of this analyst, this is not a serious problem; there are many cases in which other conservatisms are employed. As a minimum, however, it does constitute an example of incorrect documentation.

3) Use of undue optimism in assessment of credit for human redundancy.

On page 1.5-584 of the IPPSS it is stated that following an important transient there would be four people present in the control room: two control board operators, one of whom is an SRO, the watch/supervisor (an SRO), and the STA who "does not have an operating license, but has been trained in the mechanics of accident control and plant response characteristics." For certain major transients (e.g., a LOCA), the report makes the reasonable assumption that all four of these people would be present within half an hour following off-normal annunciator signals. One control board operator reads the procedures related to the transient while the other does the actual interfacing with the control boards. The IPPSS reasonably assesses a high level of dependence between the two operators. A moderate degree of dependence is assessed between them and the watch supervisor and between these three and the STA. All of these levels of dependence seem reasonable when all four people are involved in the same activity. The problem is that for some transients, all four are presumed to be involved in the details of monitoring control room indications and verifying that correct

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*Kolb, G. J., (Principal Investigator), Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One--Unit 1 Nuclear Power Plant, NUREG/CR-2787, U.S. Nuclear Regulatory Commission, Washington, D.C., 1982.

switching actions have been carried out. Thus, for several applications, including one of the two operator/small LOCA actions evaluated in Section 2.5.4 below, the IPPSS assumes that four people would have to fail, whereas this analyst would assume only three people at the most. Given an assessment of moderate dependence for the STA, the assumption of his involvement in some detail can result in a recovery factor as high as about 85 percent. If the STA would not be involved, the IPPSS assumption results in undue optimism in their assessment of credit for human redundancy.

4) Use of optimistic assessments of human performance under stress, especially for cases of multiple problems.

As in the ZPSS, perhaps the major fault in the HRA for the IPPSS is the use of more than one operator action designator (i.e., OP1 through OP5 described beginning on page 1.3-127) in the same sequence of events (ET designators) without modifying the HEPs for the added stress of less time or less practice or less familiarity with sequences involving multiple faults, e.g., loss of feedwater plus anticipated transient without scram (ATWS). The use of more than one OP designator in a sequence implies an unreasonably optimistic assumption that there would be no exacerbating effect due to the interaction of stress effects.

However, apparently none of the multiple fault transients have an impact in the systems analysis, so the above assumptions made about stress may not be important for this particular PRA. Furthermore, the assumptions in the IPPSS about stress levels of operators in responding to initiating high-pressure recirculation seem reasonable, and this is the only system event in which human error has a major impact in the systems analysis. As in the ZPSS, the IPPSS bases modifications of the stress model in NUREG/CR-1278 on the stated degree and frequency of practice in carrying out HP recirculation procedures. If their statements are valid (which this analyst presumes is the case), their modifications to the stress model seem reasonable.

Another problem in this HRA (as well as in the HRA for the Zion plant) is that the application of the IPPSS human performance models for LP or HP recirculation is sometimes made for response to events when considerably less time is available for successful operator intervention than was assumed for these two models. For example, on page 1.3-120 there is an event "K-4 Manually Deenergize and Rods Drop." This requires, according to the IPPSS analysis, that a successful manual trip of the reactor be made within 10 minutes of ATWS. As in the ZPSS, the IPPSS uses the LP Recirculation Model for human performance response to this event. The model presumes that four people are present; this is not a reasonable assumption in this analyst's opinion. Credit should <u>not</u> be given for the presence of the STA within 10 minutes of a transient initiation. Furthermore, if the analysis of the time available for manual intervention is incorrect (as it was judged to be so in the Sandia evaluation of the Zion study), and the available response time is actually only 2 minutes rather than 10, then no credit at all should be given for any operator intervention. (However, for this particular transient apparently the issue of human intervention is not as important in the systems analysis as it was in the ZPSS.)

5) Use of persons to estimate operator performance in place of simple measurements.

As in the Zion study, estimates of response times were obtained by interviewing operating personnel when it would have been possible to take actual measurements. Skilled personnel typicaly underestimate how much time it will take them to perform various tasks.

For example, Table 1.5.2.2.1-14 on page 1.5-343 entitled "Indian Point 2 Offsite Power Recovery Actions" provides "estimated action time" for several recovery actions. Some of the time estimates have very wide margins because they deal with repair of defective equipment. For cases such as these, operator estimates and records of repair time would constitute reasonable sources of information. (However, the report does not document how these estimates were obtained.) For other operator actions requiring much shorter times, actual time measures could have been taken--or at least simulated in talk-throughs and timed.

6) Lack of documentation on how expert opinion was used.

As was noted in the Sandia evaluation of the ZPSS, nowhere in that report, or in the IPPSS report, is there a description of the methods used for psychological scaling (the technology of using expert opinion). Without evidence that recognized methods were employed, it is not possible to have much confidence in data derived by the use of expert judgment. This criticism especially applies to cases in which histograms of cumulative probabilities of correct action over time were derived from expert opinion. This analyst has no confidence in the ability of operators to reliably make such multidimensional, absolute judgments.

7) Incomplete documentation of data sources used for estimated human performance.

Sufficient documentation was provided for tracing the use of estimates from NUREG/CR-1278. However, in the case of the use of expert opinion, and in some cases where the data source was not stated at all, or where a description of relevant performance shaping factors is not provided, it is not possible to evaluate the estimated HEPs in the IPPSS. There are many cases of this lack of documentation. One example is found on page 1.5-419 where it is apparently assumed that if there is a failed dc power fuse, it will be detected 100 percent of the time during the operator check of the status of the panels once per shift. Without describing fully the "operator check" each shift, one does not know whether this is merely a casual "look around the panels" as is done at some plants, or whether that particular dc power fuse is an item on a shiftly checklist, such as that employed by Arkansas Nuclear One Unit 1 personnel. If the latter is the case, there would be a high probability of detection each shift, but not 100 percent. If the former is the case, depending on the type of display, the credit allowed for the shiftly check might be very, very small.

8) Use of optimistic assessments of dependence among tasks done by the same person.

In addition to the optimistic assumptions about dependence, among team members (see item 3), the IPPSS provides (on page 1.5-123) a rule for within-person dependence that can result in optimistic assessments. The rule is for the tasks of a person successively restoring valves to their proper positions after test or maintenance. The report states, "For those routine actions, where written procedures are used, the level of dependence between the restoration of the first two valves is judged as moderate and the level of dependence for all other valves is complete."

This general rule could lead to extreme optimism for cases where the true level of dependence for the operator's errors of omission is complete. That is, for certain valve configurations (as described in Chapter 13 of NUREG/CR-1278) it is very likely that if an operator fails to restore one of two or more valves, he will always fail to restore the other(s). If for example, there are two redundant valves in a system, and if one assumes a basic error probability of .003, the application of the above IPPSS general rule would result in an estimated joint HEP of

$$.003 \times \frac{1 + 6(.003)}{7} = 4 \times 10^{-4},$$

whereas the correct estimate would be $.003 \times 1.0 = 3 \times 10^{-3}$, nearly a factor of 10 higher.

This same problem was also found in the ZPSS, and as in that study, this analyst could not find that the general rule was ever used. If it has been used, recalculations are in order.

9) Possible insufficient consideration of common-cause failures from human errors.

Insufficient documentation was provided to evaluate whether the possibilities for common-cause failures from human errors were appropriately assessed. For example, in the reactor protection system (RPS), the report mentions (on page 1.5-389) the possibility of common miscalibration errors but states that "...most calibration activities, even if performed in error, do not result in an instrument that fails to provide a trip." No further clarification is given.

10) Possible insufficient consideration of errors in restoring safety components after test, maintenance, or calibration.

It is not clear from the IPPSS if sufficient consideration was given to the possibilities for unavailability of safety components due to restoration errors after maintenance, calibration, or testing. This analyst has the impression that optimism may have occurred. But the lack of discussion in this area did not permit an accurate evaluation. For example, in addition to the short discussion above in the IPPSS in rejecting the possibility of common-cause calibration errors for the RPS, nothing is said about the possibilities for common-cause influence from failure of technicians or operators to restore circuits or components to the normal status after disruption of the normal status to permit the calibration. It may well be that logic testing provides sufficient recovery factors, but the report does not provide clarification.

11) Frequent use of conservatism in the HRA.

Apart from specific comments above on the possibility of undue optimism in the IPPSS for certain analyses, it was apparent that in several cases the PRA team did incorporate measures of conservatism in other analyses. In several cases, even though this analyst judged that some aspect of the IPPSS HRA for a given task was optimistically assessed, other aspects for the same task were treated so conservatively that this analyst's overall impression was that the final analysis was not optimistic--and even pessimistic in some cases.

The overall impression received is that those responsible for the HRA in the IPPSS attempted to avoid undue optimism in assessing the effects of human performance. Their occasional use of some inappropriate optimism (in the opinion of this analyst) reflects either honest errors of judgment in their analyses or an inappropriate evaluation of their estimates by this analyst. The latter is certainly possible in light of the lack of documentation provided in the IPPSS and this analyst's unfamiliarity with the Indian Point plants.

2.5.4 Quantitative Evaluation

Only four human error terms are shown in Section 3.0 of the Sandia evaluation to have a major impact on the systems analysis. These are related to high-pressure and low-pressure recirculation after a LOCA.

Following is an evaluation of the IPPSS HRA assessments for the four terms.

1) Failure to initiate switchover to high-pressure recirculation after a small LOCA.

On page 1.5-584, this term is designated as Q_{H1}, the failure to initiate switchover to high-pressure recirculation. The IPPSS estimates that it takes at least 2 hours, and more likely 10 hours, for HP recirculation to be needed after a small LOCA. This need should be recognized when the transient is properly diagnosed, and the time to initiate recirculation is indicated by a low level alarm in the refueling water storage tank (RWST). A well-organized crew would be monitoring the RWST level indicator and would not likely be taken by surprise when the alarm sounds.

No assessment is given in the IPPSS for the operating crew to fail to recognize they have a small LOCA. By implication, the HEP is zero. This seems a reasonable assumption; this analyst's latest model for this type of diagnostic error by the control room team 2 hours after it is recognized that something is amiss gives a nominal HEP of between 10^{-4} and 10^{-5} with an error factor of 30.* Given the fact that the Indian Point operators are well-versed on what pattern of stimuli is associated with a small LOCA, and that, as stated on page 1.5-583 of the IPPSS, the time window is 60 minutes, the failure of all four people in the control room to recognize the nature of the problem and still allow sufficient time for the switchover actions should be vanishingly low.

The actual switchover procedures should be initiated when the RWST low level annunciator comes on. Given that no misdiagnosis has been made (as stated above), there should be plenty of time for the operators to eyeball the vertical analog meter which displays RWST level. In one sense this is a dynamic task because it involves the monitoring of a constantly but slowly changing display indication. However, even if the operators get involved elsewhere and forget to monitor this display (which seems unlikely), the RWST low level

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*Swain, A. D., "Modeling of Response to Nuclear Power Plant Transients for Probabilistic Risk Assessment," Proceedings of the 8th Congress of the International Ergonomics Association, Tokyo, Aug. 1982.

annunciator offers a very good signal to tell them it is time to initiate switchover to recirculation. The effectiveness of this annunciator will depend on how many competing auditory annunciations are occuring at about the same time as the low level annunciator. The IPPSS does not provide this information.

However, the crew apparently have 60 minutes in which to initiate switchover, so there is time to recover even if they forget to monitor the low level indication and if they don't take proper notice of the related annunciator. It appears to me that an operating crew would really have to be utterly confused if the switchover procedures were not initiated within the allowable time.

The IPPSS uses the same arguments made in the ZPSS for the error of failing to initiate switchover. They use a basic HEP of .003 and double it for moderately high stress, using NUREG/CR-1278 as their guide. They assume that the omission error would be a function of all four personnel in the control room, using the assessed levels of high dependence between the two operators, and moderate dependence for the watch supervisor and the shift technical advisor. On page 1.5-586 the HEP (median) is calculated correctly as 6.6×10^{-5}.

However, if the IPPSS is correct in assessing the switchover to the recirculation phase as a dynamic task (as stated on page 1.5-580), rather than .003, the report should use .015 as the nominal HEP for this task. (The .015 is calculated from Table 20-23 in NUREG/CR-1278 as the basic HEP of .003 times 5 for dynamic tasks under moderatly high stress by highly skilled operators.) Recalculating their equations with .015 as the nominal HEP gives

$$.015 \times \frac{1.015}{2} \times \frac{1 + 6(.015)}{7} = .0012$$

This is a factor of 18 greater than the IPPSS joint HEP.

If one works out the problem in a different manner, using more detailed analysis, the joint HEP is even smaller than their 6.6×10^{-5}. Assume for example that the monitoring of the RWST level indicator is considered a dynamic task and that only the two control board operators are involved, with high dependence between them. Using the basic HEP of .003 but multiplying it by 5 (for dynamic tasks under moderately high stress) and again by .5 for the second operator (high dependence) gives a joint HEP of .0075. For the annunciator recovery factor, assume both operators and the watch supervisor are involved, with the above levels of dependence assigned to them. Also assume five alarms (i.e., four nonrelated competing alarms). The basic HEP for responding appropriately to the low level annunciator is .003 (from Table 20-24 in

NUREG/CR-1278), the second operator's HEP is .5, and the watch supervisor's HEP is about .15. The joint HEP for all three people failing to be cued by the alarm is thus 2×10^{-4}. The joint HEP for total failure is then $.0075 \times .0002 \cong 10^{-6}$, a number to which we would assign epsilon.

In discussions with IPPSS personnel on July 27 and 28, 1982, it was determined that there are annunciators for both the low level and low-low level of the RWST level indicator. Therefore, the above analysis can be taken as an approximation of the failure to initiate switchover. This estimate does not include any other human error contribution not identified by the IPPSS analysts.

2) Switch 7 turned to the "ON" position and no corrective actions are taken.

This term is identified in the IPPSS as $0.136Q_{H1}$, and is "Switch 7 is turned to the 'on' position [which stops all safety injection pumps] and no corrective actions are taken." Once the switchover initiation is begun, it is still possible for the control room personnel to make a selection error in the "eight-switch sequence" described beginning page 1.5-576. They have decided that high-pressure recirculation is required, and they use a book of procedures, with one operator reading and the other performing the switching actions. For a small LOCA, switch 6 should be operated, but switch 7 skipped. If switch 7 is erroneously selected, all safety injection (SI) pumps will be stopped.

There are several ways in which this error could be made. The operator giving the oral instructions could misread or misspeak. The second operator, given that the first operator is correct, could misselect. There is not sufficient information to make an analytical estimate of the error probability, since it would depend quite a bit on the control board design and the type of written instructions. This is clearly a static (nondynamic) procedure, and the IPPSS correctly assigned a .003 basic HEP, multiplying it by 2 for the moderately high stress level. They also reasonably state that the error would be a function only of the two control board operators--the watch supervisor or shift technical advisor would not be involved in this detail.

Given that the error was made, the IPPSS assigns a recovery factor. They use the same .006 HEP and assign it to the watch supervisor, and assume that the STA also has a chance of seeing the error, based on high dependence. It is not possible to evaluate the recovery factor because the report does not indicate what the recovery cue is. It is stated at the bottom of page 1.5-577 that "Low pressure in the SI pumps suction header is annunciated in the control room." If this means that the above error would result in an annunciation, the recovery factor should be much better than that indicated in the IPPSS. If the recovery cue really should be assessed a nominal HEP of .006, and if the STA if given no credit (which seems a reasonable assumption to this analyst) and that the shift supervisor is assigned the usual moderate level of dependence, the joint HEP is much higher than that given in the IPPSS.

Assuming that there is an annunciator as a recovery cue, it would be reasonable to give credit to three people. If it is further assumed that there are four competing annunciators (i.e., a total of five nonrelated ANNs alarming at about the same time), the joint HEP for the recovery factor is the same 2×10^{-4} calculated earlier. Thus, the total unrecovered failure probability would be the same $.006 \times \frac{1.006}{2} = .003$ probability of failure of the two operators multiplied by 2×10^{-4}, or epsilon.

If there is no annunciator, and if we assume that the watch supervisor has a moderate level of dependence for this task, his failure probability would be .15. With no recovery credit for the STA, the total failure probability would be .003 (the joint HEP for the two operators) multiplied by .15 (for the watch supervisor), or 4×10^{-4}. This is a sizeable increase from the IPPSS estimate of 6.6×10^{-5}.

In discussions with IPPSS personnel on July 27 and 28, 1982, it was determined that the above annunciator would indeed furnish a strong recovery factor as indicated in the sample analysis in this section. Therefore, the above analysis assuming the annunciation recovery factor can be taken as an approximation of the error and failure to recover from inadvertent turning of switch 7. The IPPSS analysts also determined that the operators when noting the annunciator would quickly turn back on the SI pumps, and that suction to the pumps would be available. That is, there would be no danger of burning up the pumps because of lack of suction.

3) Failure to initiate switchover to low pressure recirculation after a large LOCA.

On page 1.5-602, for the joint failure probability of the control room personnel to initiate LPR within time, several assumptions are made. It is assumed that LPR is needed 20 minutes after the large break and that the allowable time window is 20 minutes. It is assumed that all four people would be involved (the two control board operators with high dependence and the watch supervisor and shift technical advisor with moderate dependence). This assumption seems reasonable. A very high level of stress is assessed, which yields a .1 basic HEP based on the Large LOCA curve in the Handbook. Then, because the crew have had extensive simulator practice in coping with a large LOCA, this .1 is divided by 2, or a modified basic HEP of .05. This was the same correction factor applied in the ZPSS, and this analyst does not take issue with this modification.

With the above assumptions, the IPPSS joint median HEP becomes 9.1×10^{-4}, a value which is about the 30 minutes value for a team, as determined by this analyst's new model for correct diagnosis of a transient.* Using this value as the median of lognormal distribution, and using an error factor (EF) of 20, an HER of 4.75×10^{-3} is calculated. This analyst accepts the IPPSS estimate as reasonable.

4) For IP-2 (3), switch 6 (5), in addition to switch 7 (6), is turned to the "ON" position and no corrective actions are taken within the available time.

On page 1.5-603 (paragraph 2) assessment is made of the unrecovered operator error of turning switch 6 to ON which closes MOVs 746 and 747, and later turning on switch 7 which trips the SI pumps. For this error, the .1 basic HEP is used without modification, a reasonable assessment. However, the document now allows a level of low dependence for the STA and assigns recovery to all four people in the control room. For HPR, the recovery was restricted to the watch supervisor and STA and both were assigned moderate dependence. No reason is given for this change in assumptions. This analyst believes these changes for LPR may well be optimistic.

Unlike the equivalent switching error for HPR, in the LPR situation, there is no annunciator recovery factor (information obtained from discussion with IPPSS personnel). Furthermore, in the emergency procedures, there is no direction to the operators to check the flow indicators for the low head injection paths after completion of the 8-switch sequence. The only statement this analyst could find occurs as a NOTE right after step 2.2 "Recirculation Phase" in the IP-2 procedures. Part of the lengthy note says that "recirculation flow to the RCS must be maintained at all times." It is very poor practice to place an important instruction in a note.

This same note does not appear in the IP-3 emergency procedures, but there is a possible recovery about 11 steps after the 8-switch sequence is completed. This step tells the operator to check the number of Recirculation Pumps operating and whether or not both RHR

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*Swain, A. D., "Modeling of Response to Nuclear Power Transients for Probabilistic Risk Assessment," Proceedings of the 8th Congress of the International Ergonomics Association, Tokyo, August 1982.

heat exchanger flow paths are open (which include MOVs 746 and 747 which would have been closed if the switching error in question were made). It appears to this analyst that at least there is a better possibility for operator recovery of the error in the IP-3 procedures than the IP-2 procedures. Lacking any real information, however, this analyst will assume that the situation is the same in both plants. The IPPSS analysis for the switching error for the two plants are identical.

Lacking any specific instructions in the procedure to check the recirculation flow into the cold legs into the reactor vessel, reliance must be placed on the knowledge and memory of the operators to check flow. This is not an optimum method of operating under emergency conditions.

Using the IPPSS assumptions, the joint median HEP is 10^{-4}, with a derived mean HER of 5.26×10^{-4} (based on a lognormal distribution of HEPs and an EF = 20). This estimate would change materially if the equivalent assumptions from the HPR analysis were made:

1) Recovery credit for the SWS and STA only.

2) SWS and STA both moderate level of dependence.

The joint HEP of .055 would not change, but the recovery factor would be much reduced. The failure of recovery becomes:

$$\left(\frac{1+6x.1}{7}\right)^2 = .052$$

Thus, the joint unrecovered HEP would be .055 x .052 = .0029, or about a factor of 30 higher than the IPPSS estimate of 10^{-4}. Presumably, the mean HER would also be increased about a factor of 30, to about 1.5×10^{-2}.

This analyst is unable (with the information available) to ascertain whether or not the assumptions made in the IPPSS for this error are reasonable. A sensitivity analysis could determine if a factor of 10 increase in their HER would have a material impact in the system analysis.

2.5.5 Summary of the Review of the Human Reliability Analysis

This summary is very similar to this analyst's summary comments on his review of the Zion Probabilistic Safety Study. As in that review, the major problem in reviewing completely the HRA for the Indian Point Probabilistic Safety Study is the lack of documentation. While this is also a problem for the PRA as a whole, it is a much bigger problem for a review of an HRA. HRA deals with the most difficult component of a system to understand and to quantify.

Because of the lack of documentation in the IPPSS, this analyst had to interview IPPSS analysts to complete his evaluation of the only four error terms which are shown in the systems analysis to have a material impact in the PRA. These are four terms which deal with switchover to recirculation after a LOCA.

The close correspondence of the IPPSS HRA with the ZPSS HRA apparently reflects a judgment by the human reliability analysts that there is sufficient similarity in the behaviors for the tasks analyzed in both PRAs so that such extrapolation is warranted. This analyst does not have sufficient information to evaluate this generalization.

While the IPPSS does not deliberately appear to be optimistic in its assessment of human errors, assumptions made regarding the credit to be given for more than one person in the performance of several tasks did on occasion have that effect. Furthermore, the development of only two stress models (for high-pressure recirculation and for low-pressure recirculation) and the misapplication of these models to completely different situations also had the net result of probably underestimating the effects of human errors in responding to some unusual events, especially in those cases where there is more than one unusual event to contend with or when the allowable time for the control room personnel to respond is so short that it is unlikely that all four persons would be present.

The above optimism is countered, at least for some analyses, by a deliberate decision not to take full credit for certain recovery factors, and by the use of rather wide uncertainty bounds.

2.6 Estimation Methodology

2.6.1 Introduction

In this section, we examine the Indian Point Probabilistic Safety Study (IPPSS) estimates of initiating event rates and the failure probabilities and unavailabilities of components and systems. The estimation methodology is the same as that used in the Zion PSS, so the general comments made in our letter report (dated 3/5/82) to the NRC on Zion apply here. For completeness, though, many of the comments are repeated here. Our emphasis is on identifying the strengths, weaknesses, and potential effects of the methodology used. The comments thus apply to IPPSS and to other studies that may adopt the same methodology. Contributions of the methodology to specific accident sequence estimates are addressed in Section 3.

Future events, such as human errors, the failure of reactor components and systems, and the resulting consequences, cannot be foretold exactly. However, by careful modeling of the occurrence of these events as the outcome of random processes, this unpredictability can be gauged and assessed. Developing these models is an essential activity in a probabilistic risk assessment (PRA).

The numbers that go into a probability model, e.g., failure rates and probabilities, component availabilities, and human error probabilities, are not known exactly. Indeed, since they are quantities in a model, which is only an approximation to reality, the notion that they exist and are knowable, as, for example, is the case for a physical constant, such as the speed of light, is somewhat ephemeral. Nevertheless, within the context of the specified model, it is necessary to estimate these quantities. Obtaining estimates, substantiating them, and conveying the possible errors--the uncertainty--present in these estimates pose considerable problems for a risk analysis. The authors of the IPPSS, (whom we shall refer to as Indian Point) approached these problems using Bayesian methodology. Under this approach, the study team represented, probabilistically, their prior beliefs about the rates and probabilities of interest, then modified these beliefs by historical data obtained from Indian Point's experience (if available), and convoluted them to yield a probability distribution representing their posterior beliefs about the frequency and consequences of various accidents.

We undertake a limited sensitivity study which the IPPSS authors did not do. If the IPPSS estimates are to be convincing, one needs to know the assumptions made and the extent to which the results depend on them.

Bayesian methodology applied to risk assessment is also new. Readers of the IPPSS might therefore be overwhelmed, enthralled, or mystified by it, so we begin this review by making some general comments about Bayesian methodology and the IPPSS rendition of it.

2.6.2 Bayesian Methodology

Consider a component that either succeeds or fails on demand. Assume that in a sequence of n demands the result on each demand--success or failure--is independent of the results on the other demands and assume that a constant, unknown failure probability, p, underlies the sequence. That is, assume a coin-tossing model. Then the probability of observing k failures in n demands is

$$P(k;\ n,\ p) = \frac{n!}{k!(n-k)!}\, p^k(1-p)^{n-k},$$

the binomial distribution. The problem is to estimate p, given data of k failures in n demands. Conventional statistical methodology yields point estimates and confidence intervals based on this model.

The Bayesian, however, seeks to incorporate other information about p. He (the generic he) expresses his state of belief about p by a probability distribution, g(p). In principle, this distribution is specified prior to observing the data, to maintain independence, and so is called the prior distribution (Indian Point calls it the generic distribution). By Bayes' Theorem (which is a straightforward manipulation of conditional probabilities) the data are used to modify the prior distribution, the result being called the posterior distribution of p (Indian Point calls it the updated distribution). To wit,

$$g(p|k,n) = \frac{p(k;n,p)\ g(p)}{\int_0^1 p(k;n,p)\ g(p)dp}\ .$$

One then presents this distribution or selected moments and percentiles to summarize his posterior degree of belief about p.

The appeal of this analysis is that people cognizant of the component surely know more about p than just what is embodied by the data, so let's incorporate that information. A difficulty is in determining g(p). One has to translate his knowledge and beliefs to probability. He has to say, "What I know about p is equivalent to knowing that it was generated at random from g(p)." This translation is difficult and fraught with peril. Whether one can justify such precision is open to question. Also, one can question whether the updated quanitified beliefs of some person or persons are of much value to those who may not share those beliefs. In the following sections, we examine how Indian Point handled these difficulties. First, though, some comments about terminology.

In the preceding and subsequent discussions, we use the term "probability," as a parameter in a model, e.g., the parameter p above, or a parameter calculated from a model, such as the probability of no failures in τ hours, given the constant failure rate model which parameter λ. One can think of a model as a mathematical representation of what would happen in infinite repetitions of some hypothetical experiment, but that's not a requirement. We use the term "personal probability," or "Indian Point's probability," to denote probabilities calculated to reflect degree of belief. We also distinguish between failure rates, which are dimensioned failures per unit time, and failure probabilities, which are dimensionless.

Indian Point calls both of the latter "frequencies," and define these as the outcome of an experiment involving repeated trials, either an actual experiment or a "thought experiment" (p. 0.4-1). Thus, rates and probabilities are not distinguished (so we see a "probability" of 4.11 on p. 1.5-161), nor are estimates of probabilities or rates, which result from a finite number of repeated trials, distinguished from the parameter being estimated, which correspond to infinite repetitions. Indian Point uses "probability" variously as quantified degree of belief, confidence, or knowledge (which are not all the same). In the following sections, we consider the estimation of component failure rates and probabilities, initiating event rates, and maintenance unavailability, and then combining these estimates to estimate system failure probabilities.

2.6.3 Treatment of Component Failure Data

Indian Point's estimates of component failure rates and probabilities were obtained from the following sources:

- Indian Point site-specific experience, as given by LERs and other station records

- Industry-wide LER summaries on valves, pumps, and diesel generators published by EG&G.

- WASH-1400

- IEEE-500 estimates of electrical component failure rates and probabilities

The last three sources were used to develop prior distributions, which were then modified by the Indian Point data, using Indian Point's DPD (discrete probability distribution) arithmetic, to arrive at the posteriors. The means and variances of these distributions are reported in the IPPSS Tables 1.5.1-4 and 1.6.1-4.

From the authors' Bayesian orientation one would expect their prior probability distributions, regardless of how they are developed, to be described only as their prior degree of belief about the unknown Indian Point parameters. But they make the much stronger claim (p. 0.14.3) that these are "frequency distributions," the "known results of experiments on populations." They are said to represent the "variation of performance of individual components within the population." This is a presumptuous claim and unnecessary from the Bayesian viewpoint. It is unclear why Indian Point made it. They contradicted this claim when they subsequently assumed that individual components of a given type, e.g., all motor-operated valves at Indian Point 2, all have the same constant failure rate, rather than individually different rates.

Most of Indian Point's prior distributions are based in part on WASH-1400. It is not at all clear from WASH-1400 how the lognormal distributions given there are to be interpreted, but there is no basis to regard them as the results of (infinite) "experiments on populations." In fact, the nuclear plant data in WASH-1400 amount to one year's worth (1972) of (what are now called) LERs. For Indian Point to regard the distributions supplied by WASH-1400, even after they are stretched out so that the 5th and 95th percentiles become the 20th and 80th, as known frequency distributions, and to call them "generic" is unwarranted.

One consequence of assuming that Indian Point's prior distributions are the frequency distributions of plant-to-plant variability is that in order to proceed with the derivation of the posterior distribution you must next assume that the Indian Point units are random samples from the population of plants. This, too, seems difficult to support.

What seems most plausible is to regard Indian Plant's prior distributions as their representation of their prior personal belief, or knowledge, of the failure rates and demand probabilities for classes of components at Indian Plant. These priors, rather than being obtained by careful introspection and elicitation of the knowledge possessed by the study team or the Indian Point personnel, as one would expect Bayesians to do, were obtained by applying ad hoc prescriptions to the numerical results published in the above sources. As we shall see, the effect of this approach is quite uneven. Also, as we shall see in our Sections 2.6.6 and 3, there are important, unannounced exceptions to Indian Point's treatment of WASH-1400's 5th and 95th percentiles as 20th and 80th.

Regardless of whether one accepts, rejects, or ignores the claims made by Indian Point for their prior distributions, the important question remains as to what effect these distributions had on their estimates. Just looking at the data tables doesn't tell you. In fact, the lognormal distributions identified as "generic" priors are not even used in Indian Point's calculations. The actual prior distributions used are discretized versions of these distributions. Just how the discretization is done is not described. Nor are the discrete priors ever provided in the report (which means it is impossible to verify any of the posterior distributions). This might be a minor point except that in some of the systems analyses it was found that discretizing a distribution could considerably reduce its variance.

In order to identify the contributions of Indian Point's priors to their results, we pretend the "updated results" are based on a statistical (as opposed to Bayesian) analysis. In a statistical analysis, given data consisting of f failures in T hours and assuming a constant failure rate, one would estimate that failure rate by $\lambda^* = f/T$, where the asterisk denotes an estimate. Under the assumption that T is fixed and known, the variance of λ^* would be estimated by $var^*(\lambda^*) = f/T^2$. Indian Point provides a posterior mean (their point estimate) and variance. If we equate these to f/T and f/T^2, respectively, and solve for f and T, then we obtain pseudo-data effectively corresponding to the information assumed by Indian point in estimating a failure rate. Alternatively, one can do a Bayesian analysis beginning with some uninformative or "flat" prior distribution, then modify it by f and T to obtain a posterior distribution which would have (at least approximately) a mean and variance equal to f/T and f/T^2. Also, this correspondence between f/T and the posterior mean is consistent with Indian Point's practice of equating the value of f/T in the EG&G reports to their prior mean, so we are not doing anything funny by this transformation. If Indian Point had followed conventional Bayesian practice by choosing a "natural conjugate" prior distribution, in this case a gamma distribution, then the parameters of the posterior distribution, which, fortunately, is also a gamma distribution, are directly interpretable as effective data--number of failures and number of hours. Indian Point used discretized lognormal distributions for their prior distributions, so we can't make this correspondence exactly. But, and this is one saving feature of a Bayesian analysis, with enough data the prior distribution doesn't matter too much, so approximating a discretized lognormal distribution by a gamma distribution should be reasonably adequate.

Thus, the failure rate posterior means and variances in the IPPSS Tables 1.5.1-4 and 1.6.1-4 can be converted to effective data, say f_{POST} failures in T_{POST} hours. The Indian Point-specific f and T are given, so we can subtract them from the posterior effective f and T to determine the effective f and T associated with the prior distribution:

$$f_{PRIOR} = f_{POST} - f_{IP}$$

$$T_{PRIOR} = T_{POST} - T_{IP}$$

For example, consider the first entry in IPPSS Table 1.5.1-4. The posterior mean and variance, labeled "Updated," are 7.40(-8)/hr and 5.89(-15)/hr^2. Equating these to f/T and f/T^2 yields

$$T_{POST} = \frac{7.40(-8)}{5.89(-15)} = 1.26(7) \text{ hrs.}$$

$$f_{POST} = 7.40(-6) \times 1.26(7) = .9$$

That is, Indian Point's posterior mean and variance correspond to what one would estimate given only the data of .9 failures in 12.6 million hours (mhrs). The Indian Point experience consists of zero failures in 6.0 mhrs. Thus, the difference, which is Indian Point's rendering of the non-Indian Point information, amounts to .9 failure in 6.6 mhrs. (We note in passing that expressing prior information as being equivalent to .9 failures in 6.6 mhrs. is more scrutable than being told it is equivalent to a lognormal distribution with a 20th percentile of 2.8(-8)/hr. and an 80th percentile of 2.8(-7)/hr.)

From the Indian Point data alone, the upper 95 percent statistical confidence limit on the underlying failure rate would be 5.0(-7)/hr. From the effective posterior data, the upper 95 percent statistical confidence limit is 3.8(-7)/hr., so in this case, and from this view, the prior does not have a marked effect.

For demand probabilities, given data of f failures in n demands, one would obtain the estimate, $p^* = f/n$, and the estimated variance, $var^*(p^*) = p^*(1-p^*)/n$. These can be equated to Indian Point's posterior mean and variance to solve for an effective f and n. For small p^*, these solutions correspond to those for λ^* with n replacing T.

Table 2.6-1 gives the effective prior data for all the entries in the IPPSS Tables 1.5.1-4 and 1.6.1-4. The contributions of the priors to the final results vary considerably. In many cases, the prior denominator, n or T, is roughly the same size as that for the Indian Point data, e.g., Indian Point-2 components 1, 5, 8, so the effect is roughly to decrease the variance by a factor of two. The precise effect depends on the numerator. In several cases, the prior leads to a smaller and more precise estimate than would be obtained from the Indian Point data alone by effectively subtracting from the numerator while adding to the denominator (components 4, 11, and 20 for IP-2; components 11, 14, 21, and 29 for IP-3). In other cases (including components 2, 7, 10, 16, 18, 19, 22, and and 34 at both units) the prior denominator is roughly ten or more times that for the plant-specific data alone, so considerable additional precision is imparted. One case that stands out is component 35, IP-3, where the prior effectively amounts to 712.7 failures in 52.9 x 10^6 hrs. This is probably due to a typo in the positive variance. There are three cases (component 17 at both units, component 13 at IP-3) where the prior leads to less precision than the Indian Point data alone would by subtracting from both numerator and denominator. Whether or not the contributions of the prior distributions are fair and just, depends on the actual information contained in the source documents. Whether this question is worth worrying about in the IPPSS depends on where the various component events occur in the system models. We address this question in Section 3.

It should be noted that the preceding analysis, and Indian Point's, is predicated on the Indian Point data given in the report. We have no way of validating the data, of determining the accuracy of the reported numerators and denominators. Sections 1.5.1 and 1.6.1 of the Indian Point study indicate a good deal of care in collecting component data.

Table 2.6-1

Plant-Specific and Effective Posterior and Prior Data

INDIAN POINT 2

COMP	NO. OF FAILURES	SERVICE HRS OR DEMANDS		UPDATED MEAN	UPDATED VARIANCE	EFFECTIVE POSTERIOR		EFFECTIVE PRIOR	
1	0.	6.000E+06	H	7.400E-08	5.850E-15	.9	1.256E+07	.9	6.564E+06
2	0.	7.030E+02	H	1.990E-08	1.630E-13	.0	1.067E+05	.0	1.060E+05
3	0.	1.444E+03	D	7.020E-05	1.090E-08	.3	6.440E+03	.5	4.996E+03
4	1.	4.440E+05	H	9.200E-07	1.030E-12	.8	6.522E+05	-.2	4.492E+05
5	1.	3.010E+05	H	2.560E-06	4.810E-12	1.4	5.322E+05	.4	2.312E+05
6	3.	1.261E+03	D	2.320E-03	1.190E-06	4.5	1.950E+03	1.5	6.886E+02
7	0.	8.160E+02	H	9.650E-09	2.750E-12	.0	2.573E+04	.0	2.492E+04
8	1.	1.050E+03	D	7.490E-04	4.010E-07	1.4	1.868E+03	.4	6.175E+02
9	0.	4.440E+05	H	1.710E-07	7.290E-14	.4	2.346E+06	.4	1.902E+06
10	0.	3.700E+04	H	7.300E-08	2.830E-13	.0	2.560E+05	.0	2.213E+05
11	7.	7.930E+02	D	6.410E-03	7.760E-06	5.3	8.229E+02	-1.7	3.091E+01
12	0.	1.700E+01	D	2.010E-03	2.500E-05	.2	8.040E+01	.2	5.340E+01
13	0.	8.400E+01	H	1.590E-05	1.610E-08	.0	9.876E+02	.0	9.036E+02
14	0.	6.500E+01	H	1.630E-05	1.670E-08	.0	8.716E+02	.0	8.066E+02
15	0.	7.400E+04	H	2.760E-06	1.590E-11	.5	1.726E+05	.5	9.956E+04
16	0.	4.300E+01	H	1.680E-05	2.760E-08	.0	6.087E+02	.0	5.657E+02
17	2.	7.400E+04	H	1.520E-05	2.090E-10	1.1	7.273E+04	-.9	-1.274E+03
18	0.	2.000E+00	H	1.950E-0[illegible]	1.600E-07	.0	1.219E+02	.2	1.185E+02
19	0.	5.400E+01	H	1.650E-05	2.180E-08	.0	7.569E+02	.2	7.029E+02
20	1.	5.120E+03	H	6.540E-05	1.050E-08	.4	6.228E+03	-.6	1.105E+03
21	0.	6.400E+02	H	1.150E-05	2.990E-09	.0	2.862E+02	.0	2.242E+03
22	0.	3.500E+01	D	7.260E-04	1.820E-06	.3	3.969E+02	.3	3.639E+02
23	2.	7.400E+04	H	4.060E-05	2.600E-10	4.6	1.123E+05	2.6	3.933E+04
24	0.	1.480E+05	H	8.420E-07	1.540E-12	.4	4.340E+05	.4	2.860E+05
27	4.	4.240E+02	D	1.290E-02	4.150E-05	4.0	3.106E+02	.6	-1.132E+02
26	0.	2.040E+02	H	9.370E-04	2.370E-06	.3	2.778E+02	.3	7.376E+01
29	0.	2.960E+02	D	2.460E-05	1.840E-08	.0	1.327E+02	.0	1.041E+03
30	0.	2.960E+02	D	6.360E-04	2.140E-06	.2	2.972E+02	.2	1.196E+00
31	0.	4.440E+05	H	4.810E-07	8.710E-12	.3	5.522E+05	.3	1.062E+05
32	0.	1.850E+05	H	7.630E-07	1.140E-12	.5	6.693E+05	.5	4.643E+05
33	1.	9.580E+04	H	1.550E-05	2.200E-10	1.1	7.045E+04	.1	-2.535E+04
34	0.	9.580E+04	H	8.210E-06	5.820E-14	.1	1.411E+06	.1	1.315E+06
35	0.	9.540E+04	H	2.010E-06	2.300E-11	.2	8.735E+04	.2	-6.409E+03
36	1.	7.500E+01	D	2.710E-06	6.910E-12	1.1	3.922E+05	.1	3.421E+05
37	0.	1.030E+06	H	2.800E-0[illegible]	6.930E-15	.1	4.040E+06	.1	3.010E+06
39	0.	0.	H	3.220E-06	6.960E-11	.1	3.594E+04	.1	3.594E+04
40	0.	0.	H	7.520E-06	4.880E-10	.1	1.541E+04	.1	1.541E+04
41	0.	0.	D	6.280E-05	2.490E-11	1.6	2.522E+05	1.6	2.522E+05
42	0.	0.	H	8.600E-09	6.000E-15	.0	1.433E+06	.0	1.433E+06
43	0.	0.	H	8.600E-1	6.000E-17	.0	1.433E+07	.0	1.433E+07
44	0.	0.	H	8.480E-10	5.100E-17	.0	1.663E+07	.0	1.663E+07
45	0.	4.400E+02	D	4.670E-04	8.510E-07	.3	5.488E+02	.3	1.088E+02
46	0.	0.	H	8.320E-07	1.080E-09	.0	7.704E+02	.0	7.704E+02
47	0.	0.	D	1.150E-05	2.380E-09	.0	3.402E+03	.0	3.402E+03
48	0.	0.	H	2.430E-07	2.260E-12	.2	7.454E+05	.2	7.454E+05
49	0.	0.	D	3.660E-07	1.470E-13	1.0	2.639E+06	1.0	2.639E+06
50	0.	0.	H	1.660E-06	6.280E-12	.4	2.643E+05	.4	2.643E+05
51	0.	0.	H	4.280E-07	2.360E-10	.0	1.274E+03	.0	1.274E+03

Table 2.6-1 (Cont.)

Plant-Specific and Effective Posterior and Prior Data

INDIAN POINT 3

COMP	NO. OF FAILURES	SERVICE HRS OR DEMANDS		UPDATED MEAN	UPDATED VARIANCE	EFFECTIVE POSTERIOR		EFFECTIVE PRIOR	
1	0.	3.700E+06	H	9.150E-08	1.010E-14	.8	9.059E+06	.8	5.359E+06
2	0.	1.970E+02	H	2.000E-08	1.890E-13	.0	1.058E+05	.0	1.056E+05
3	0.	1.550E+03	D	6.910E-05	1.030E-08	.5	6.709E+03	.5	5.159E+03
4	0.	8.640E+05	H	2.580E-07	1.000E-13	.7	2.580E+06	.7	1.716E+06
5	1.	1.600E+05	H	3.660E-06	1.280E-11	1.0	2.859E+05	.0	1.259E+05
6	0.	2.440E+02	D	1.510E-03	2.640E-06	.9	5.720E+02	.9	3.280E+02
7	0.	2.900E+02	H	9.870E-08	4.360E-12	.0	2.253E+04	.0	2.224E+04
8	0.	3.880E+02	D	4.980E-04	4.030E-07	.6	1.236E+03	.6	8.477E+02
9	0.	4.800E+05	H	1.690E-07	6.900E-14	.4	2.449E+06	.4	1.969E+06
10	-0.	2.400E+04	H	7.700E-08	3.470E-13	.0	2.219E+05	.0	1.979E+05
11	2.	8.000E+02	D	1.360E-03	1.220E-06	1.5	1.115E+03	-.5	3.148E+02
12	0.	4.000E+01	D	1.650E-03	1.030E-05	.3	1.602E+02	.3	1.202E+02
13	1.	4.000E+01	H	1.790E-03	4.770E-05	.1	3.746E+01	-.9	-2.541E+00
14	2.	8.000E+03	H	1.500E-04	1.740E-08	1.3	8.619E+03	-.7	6.194E+02
15	0.	4.800E+04	H	3.260E-06	2.470E-11	.4	1.320E+05	.4	9.398E+04
16	0.	5.300E+01	H	1.650E-05	2.220E-08	.6	7.432E+02	.0	6.902E+02
17	3.	4.800E+04	H	4.680E-05	1.070E-09	2.0	4.374E+04	-1.0	-4.264E+03
18	0.	2.000E+00	H	1.960E-05	1.700E-07	.0	1.153E+02	.0	1.133E+02
19	0.	2.000E+01	H	1.770E-05	6.440E-09	.0	2.748E+02	.0	2.548E+02
20	0.	1.200E+03	H	9.990E-06	1.980E-09	.1	5.045E+03	.1	3.845E+03
21	1.	[illegible].000E+02	H	3.770E-04	3.300E-07	.3	7.111E+02	-.7	1.111E+02
22	0.	1.000E+01	D	7.800E-04	2.560E-06	.2	3.047E+02	.2	2.947E+02
23	0.	4.[illegible]0E+04	H	9.790E-06	2.230E-10	.4	4.390E+04	.4	-4.099E+03
24	0.	9.60[illegible]E+04	H	9.730E-07	3.340E-12	.3	2.913E+05	.3	1.953E+05
27	2.	1.8[illegible]E+02	D	1.440E-02	5.120E-05	4.1	2.813E+02	2.1	9.625E+01
28	0.	2.0[illegible]0E+02	H	9.370E-04	3.370E-06	.3	2.773E+02	.3	7.379E+01
29	1.	1.420E+02	D	1.330E-03	5.530E-06	.3	2.368E+02	-.7	9.678E+01
30	0.	1.420E+02	D	1.450E-03	1.120E-05	.2	1.295E+02	.2	-1.254E+01
31	1.	2.880E+05	H	2.670E-06	3.210E-12	2.2	8.318E+05	1.2	5.438E+05
32	0.	1.280E+05	H	8.390E-07	1.570E-12	.4	5.344E+05	.4	4.064E+05
33	0.	7.200E+04	H	3.770E-06	6.920E-11	.2	5.448E+04	.2	-1.752E+04
34	0.	7.200E+04	H	8.350E-08	6.440E-14	.1	1.297E+06	.1	1.225E+06
35	2.	7.200E+04	H	1.350E-05	2.550E-13	714.7	5.294E+07	712.7	5.287E+07
36	0.	0.	H	8.320E-07	1.030E-09	.0	7.704E+02	.0	7.704E+02
37	0.	0.	D	1.150E-05	3.380E-09	.0	3.402E+03	.0	3.402E+03
38	0.	0.	H	2.430E-07	3.260E-13	.2	7.454E+05	.2	7.454E+05
39	0.	0.	D	3.880E-07	1.470E-13	1.0	2.639E+06	1.0	2.639E+06
40	0.	0.	H	1.660E-06	6.280E-12	.4	2.643E+05	.4	2.643E+05
41	0.	0.	H	4.280E-07	3.360E-10	.0	1.274E+03	.0	1.274E+03
42	0.	5.450E+05	H	3.250E-08	1.270E-14	.1	2.559E+06	.1	2.014E+06
44	0.	0.	H	3.220E-06	8.960E-11	.1	3.594E+04	.1	3.594E+04
45	0.	0.	H	7.520E-06	4.880E-10	.1	1.541E+04	.1	1.541E+04
46	0.	0.	D	6.280E-06	2.490E-11	1.6	2.522E+05	1.6	2.522E+05
47	0.	0.	H	8.600E-09	6.000E-15	.0	1.433E+06	.0	1.433E+06
48	0.	0.	H	8.600E-10	6.000E-17	.0	1.433E+07	.0	1.433E+07
49	0.	0.	H	8.480E-10	5.100E-17	.0	1.663E+07	.0	1.663E+07
50	0.	1.440E+02	D	1.170E-03	8.963E-05	.0	1.321E+01	.0	-1.308E+02

The IPPSS analysis is also based on the assumption of constant (across time and similar components) failure rates and probabilities. This is standard in risk assessments, but the reader should be aware that it may be the source of substantial errors that are not quantifiable except by Bayesian extremists (and Indian Point doesn't go that far). Aging effects may be present and failures may cluster due to imperfect repair. Modeling such effects can be difficult and is often impossible to do with meaningful precision because of limited data. The result of the Indian Point study is not "the risk" from the Indian Point plant, but is an estimate of the Indian Point risk--an estimate built from a variety of simplifying assumptions and models.

2.6.4 Estimation of Initiating Event Rates

The initiating event frequency data for all PWRs are given in IPPSS Table 1.5.1-32 (p. 1.5-148). The basic source of their data is EPRI NP-801, modified by the data obtained from detailed examinations of the Indian Point and Zion plant records. In examining their data, we noted some differences for Indian Point in this report and the data given in the Zion study. For example, the ZPSS shows 39 and 8 turbine trips at IP2 and 3, respectively; the IPPSS shows 32 and 4. The detailed examination of Indian Point records followed the Zion study and yielded different results from EPRI NP-801. The effect, though, should be small since the Indian Point estimates, particularly for those events that have frequently occurred, are dominated by Indian Point data. Nevertheless, a detailed study of initiating event occurrences industry-wide would be of some interest. Also, we noted that in some cases, the IPPSS listings of initiating events do not match the numbers in the summary tables. For example, at IP-3, three turbine trip/loss of offsite power events are listed; only one was counted in their calculations.

The method used (but not described) by Indian Point to estimate initiating event rates is to suppose that each PWR has its own constant occurrence rate and the rates vary randomly among PWRs according to a lognormal distribution. They assume a prior distribution over a grid of (μ,σ) values--the parameters that identify a lognormal distribution--then update it by the ensemble of PWR data to obtain their posterior distribution of occurrence rates. This distribution, after discretization, serves as their prior distribution which is then "updated" by the Indian Point data (units 2 and 3 being analyzed separately).

These "generic" priors are different in the IPPSS from what they were in the ZPSS, and not just for reasons given in the preceding paragraph. In principle, they should be the same because the data and state of knowledge are the same. But consider the large LOCA initiating event. In the ZPSS, the occurrence rate had a prior mean of 1.0(-3) and a variance of 6.4(-6); in IPPSS they are 2.6(-3) and 1.8(-4). These same results pertain to all initiating events that have not yet occurred. One wonders what was learned about these events between the two studies to warrant this injection of pessimism. It turns out (from conversations with the authors) that the answer is nothing. The difference is just due to different choices of a (μ,σ) grid, guided by two analysts' concepts of what looked right at the end of the analysis. The effect is not trivial. Indian Point is estimated to have large LOCAs (roughly) twice as frequently as Zion.

As in the previous section, we can gauge the impact of the chosen prior distributions, after discretization by calculating the effective posterior data from Indian Point's posterior means and variances. The IPPSS also gives percentiles from their posterior distributions. An alternative way to express their results as effective data is to let f be the observed number of occurrences of a particular initiating event at Indian Point, then find the value of T (in years) such that the upper 95 percent statistical confidence limit on the occurrence rate is equal to Indian Point's posterior 95th percentile. For example, for large LOCA (and the other nonoccurring events), $f = 0$ and the 95th posterior percentile, λ_{95}, is 6.30×10^{-3}. The effective T is given by

$$T = \frac{\chi^2(2f+2,.95)}{2\lambda_{95}} \quad ,$$

where $\chi^2(m,\gamma)$ is the $100\,\gamma$ th percentile on the chi-squared distribution with m degrees of freedom. For large LOCA,

$$T = \frac{5.99}{2 \times 6.3 \times 10^{-3}} = 475 \text{ yrs.}$$

The Indian Point 2 experience is zero occurrences in 5 years, so the prior effectively adds on 470 LOCA-free years. Note that the total PWR experience used in the IPPSS data base is 131 years, so the assumed prior "state-of-knowledge" is effectively 339 LOCA-free years. (For Zion, the 95th posterior percentile corresponded to 0/844 years.)

The posterior mean and variance for large LOCA yield effective data of .04/21. Note though that data of 0/21 would yield an upper 95 percent statistical confidence limit of .14 occurrences per year, which is considerably more pessimistic than Indian Point's 95th percentile. The calculation in the previous paragraph better conveys the information assumed in Indian Point's analysis. The calculation of effective data from the posterior mean and variance, when it yields small fractional occurrences, may not accurately reflect the information injected by the prior distribution.

Table 2.6-2 gives the effective posterior initiating event data calculated from Indian Point's posterior 95th percentile and their posterior means and variances. Note that in all cases Indian Point's 95th percentile is more optimistic than the data alone would yield: the effective T exceeds the observed T, considerably for nonoccurring events, negligibly for those that have occurred often at Indian Point.

An assumption underlying Indian Point's analysis here, as in their analysis of component failure data, is that of a constant occurrence rate across time. No analysis is given to support this assumption, though the referenced source of transient data (EPRI NP-801) should permit such an analysis. There may be aging trends that need to be considered for transients such as steam generator tube rupture.

2.6.5 The Treatment of Maintenance Data

Indian Point models the unavailability of a component due to maintenance as the rate at which maintenance actions occur (actions per component hour, excluding cold shutdown hours) times the mean duration of a maintenance. Prior distributions for both are developed, modified by the Indian Point data to yield posterior distributions, then the distribution of the product is obtained.

Table 2.6-3 provides a comparison of unavailability estimates (including estimated maintenance frequency and average duration and average duration) using the Indian Point posterior means and using just the reported maintenance data. Only for the turbine-driven AFWS pumps do the posterior estimates appear optimistic, relative to the raw data, and then by a factor of two to three. The largest difference in the other direction is for Indian Point 2, component cooling water pump 21, but only one maintenance action has occurred. Those unavailabilities that are important in selected accident sequences will be examined further in later sections.

Table 2.6-2

Indian Point Observed and Effective Posterior Initiating Event Data; Table Entries Are (No. of Occurrences)/(No. of Yrs.)

Indian Point 2

Initiating Event Category		Plant Data	Effective Posterior From 95th Pct.	Effective Posterior From Mean, Var.
1.	Large LOCA	0/5	0/475	.04/21
2.	Medium LOCA	0/5	0/475	.04/21
3.	Small LOCA	0/5	0/57	.5/28
4.	S/G Tube Rupture	0/5	0/32	.3/12
5.	Steam Break Inside Cont.	0/5	0/475	.04/21
6.	Steam Break Outside Cont.	0/5	0/475	.04/21
7.	Loss of Feedwater Flow	35/5	35/5.6	39/5.8
8.	Closure of One MSIV	7/5	7/6.6	6.7/5.4
9.	Loss of Primary Flow	0/5	0/9	1.5/11
10.	Core Power Increase	0/5	0/44	.4/16
11a.	Turbine Trip	39/5	39/5.6	38/5.2
11b.	T. T., Loss of Offsite Power	1/6	1/10.4	1.8/8.7
11c.	T. T., Loss of Serv. Water	0/5	0/475	.04/21
12a.	Reactor Trip	36/5	36/5.6	38/5.5
12b.	Reactor Trip, Loss of Cooling Water	0/5	0/475	.04/21

Indian Point 3

Initiating Event Category		Plant Data	Effective Posterior From 95th Pct.	Effective Posterior From Mean, Var.
1.	Large LOCA	0/3	0/450	.04/18
2.	Medium LOCA	0/3	0/450	.04/18
3.	Small LOCA	0/3	0/55	.4/18
4.	S/G Tube Rupture	0/3	0/30	.3/8.0
5.	Steam Break Inside Cont.	0/3	0/450	.04/18
6.	Steam Break Outside Cont.	0/3	0/450	.04/18
7.	Loss of Feedwater Flow	12/3	12/3.5	12.4/3.3
8.	Closure of One MSIV	0/3	0/10	.5/5.2
9.	Loss of Primary Flow	0/3	0/7	1.4/8.2
10.	Core Power Increase	0/3	0/37	.3/11
11a.	Turbine Trip	8/3	8/3.5	9.6/3.5
11b.	T. T., Loss of Offsite Power	1/3	1/8.2	1.5/5.8
11c.	T. T., Loss of Serv. Water	0/3	0/450	.04/18
12a.	Reactor Trip	8/3	8/3.5	11/3.8
12b.	Reactor Trip, Loss of Cooling Water	0/3	0/450	.04/18

Table 2.6-3

Comparison of Unavailability (Due to Maintenance) Estimated Means

Indian Point 2

Components	Posterior Freq. Events/ Serv. Hr.	Posterior Dur. (hrs)	Posterior Unavail	Plant Data Freq. Events/ Serv. Hr.	Plant Data Dur. (hrs)	Plant Data Unavail	n*
Turbine-Driven AFWS Pumps	1.9(-4)	24	4.6(-3)	3.4(-4)	40	1.4(-2)	6
Motor-Driven AFWS Pumps	8.6(-5)	26	2.3(-3)	5.6(-5)	46	2.6(-3)	2
Comp. Cool. Pump 21	1.3(-4)	11	1.4(-3)	5.6(-5)	1	5.6(-5)	1
Comp. Cool. Pumps 22, 23	1.4(-4)	306	4.2(-2)	8.4(-5)	406	3.4(-2)	3
Cont. Spray Pumps	8.3(-5)	10	8.1(-4)	8.4(-5)	5	4.2(-4)	3
RHR Pumps	8.3(-5)	12	9.7(-4)	8.4(-5)	12	1.0(-3)	3
Safety Inj. Pumps	9.6(-5)	12	1.2(-3)	1.1(-4)	17	1.9(-3)	6
Serv. Water Pumps	3.3(-4)	213	7.0(-2)	3.5(-4)	254	8.8(-2)	37
Fan Coolers	8.7(-5)	16	1.4(-3)	5.6(-5)	31	1.7(-3)	5
Diesel Gens.	9.1(-4)	33	3.0(-2)	9.9(-4)	29	2.9(-2)	53
Aux. Comp. Cool. Pumps	5.8(-5)	10	5.9(-4)	No Maintenance Events			

Indian Point 3

Components	Posterior Freq. Events/ Serv. Hr.	Posterior Dur. (hrs)	Posterior Unavail	Plant Data Freq. Events/ Serv. Hr.	Plant Data Dur. (hrs)	Plant Data Unavail	n*
Turbine-Driven AFWS Pumps	1.6(-4)	25	4.2(-3)	2.5(-4)	36	8.9(-3)	5
Motor-Driven AFWS Pumps	1.7(-4)	23	4.0(-3)	2.0(-4)	30	6.1(-3)	8
Comp. Cool. Pumps	8.4(-5)	220	1.8(-2)	3.3(-5)	147	4.9(-3)	2
Cont. Spray Pumps	7.1(-5)	10	7.3(-4)	5.0(-5)	10	5.0(-4)	2
RHR Pumps	6.3(-5)	12	7.6(-4)	2.5(-5)	16	4.0(-4)	1
Safety Inj. Pumps	5.5(-5)	15	8.1(-4)	1.7(-5)	66	1.1(-3)	1
Serv. Water Pumps	3.2(-4)	46	1.5(-2)	3.3(-4)	60	2.0(-2)	40
Diesel Gens.	2.9(-4)	37	1.1(-2)	3.2(-4)	28	8.9(-3)	19
Aux. Comp. Cool. Pumps	4.4(-5)	44	1.9(-3)	No Maintenance Events			
Fan Coolers	5.1(-5)	11	5.5(-4)	No Maintenance Events			

2.6.6 Data-Free Estimates

As discussed in 2.6.3 above, to obtain prior distributions Indian Point either equated WASH-1400 5th and 95th percentiles to their 20th and 80th, or they took the ratio of WASH-1400's 5/95 percentiles as their 20/80 ratio. This can result in quite skewed and elongated distributions for which the mean and variance do not provide a very good description. Fortunately, the amount of data available from Indian Point and the DPD arithmetic can effectively chop off these long tails in the most extreme cases. There are, however, numerous probabilities and rates for which no data are available. Most of these pertain to human errors, but some pertain to hardware failures. With respect to the latter, we have encountered some instances in which Indian Point accepted WASH-1400 bounds as their own 5th and 95th percentiles, rather than stretch them out to 20th and 80th percentiles as they did in those cases in which data were available. These are:

- Rupture of a motor-operated valve. As discussed in Section 3.2.15, rupture of two MOVs leads to an interfacing systems LOCA and one of the more serious releases. If Indian Point had stretched out the WASH-1400 bounds, the estimated probability of this event would increase by five orders of magnitude.

- Pressure vessel rupture. By citing WASH-1400 bounds on the occurrence rate of this event, Indian Point dismissed it as a potential LOCA. If they had stretched these bounds, the contribution would not have been negligible.

- Pipe rupture. For pipes exceeding 3" diameter, the WASH-1400 bounds are 3(-12) and 3(-9) pipe failures per hr. Equating these to lognormal 5th and 95th percentiles yields a mean of 8.6(-10)/hr. Equating these to the 20th and 80th percentiles yields a mean of 4.5(-7), an increase by a factor of 500. Thus, for example, in the IP-2 service water system the IPPSS identifies 30 piping sections and thus estimates the failure probability as 2.58(-8) over a 1-hour period. If they had used 20th and 80th percentile assumptions, this probability would have been estimated as 1.4(-5).

The point of this discussion is not to claim one estimate is right, the other wrong, or is it to insist that Indian Point should have been consistent in their treatment of WASH-1400 bounds. As Bayesians they can specify any prior distributions they feel represents their state of knowledge. One wishes, though, the reader would be told why in some cases WASH-1400 bounds are OK and why in others they should be stretched out. The main point of these examples is that the results can be quite sensitive to what would seem to be minor differences in assumptions. This point is more than academic because of the dominant role of the interfacing systems LOCA in estimating risk.

As noted above, the DPD can chop off the tails of highly skewed lognormal (or other) distributions. Unfortunately, nothing is said in the IPPSS about the rationale for any particular discretization--how many and which discrete values were chosen. The effect can be nonnegligible. For example, the low pressure recirculation system (IPPSS p. 1.5-606 for IP-2) model contains a first order term, $1.111Q_{HI}$, where Q_{HI} is a human error term. This term is added to various other terms (treated as independent random variables) to yield the system failure probability, denoted by $Q_{LOW\ HEAD}$. The stated variance of Q_{HI} is 6.0(-4). Thus the variance of $Q_{LOW\ HEAD}$ should exceed $(1.111)^2 \times 6.0(-4) = 7.4(-4)$. The DPD convolution for $Q_{LOW\ HEAD}$, however, yields a variance of 1.4(-4). In effect, here DPD is like having five times as much "data."

2.6.7 System Quantification

Define a system as a specified arrangement of components. By a fault tree, or a reliability block diagram, a mathematical model can be developed which expresses the system failure probability as a function of the component failure probabilities and rates. Given posterior probability distributions for these component parameters, and prior distributions where no data are available, the resulting posterior distribution of the system failure probability can then be derived or approximated. The approximation method used by Indian Point is their DPD arithmetic.

In Section 3, we consider the results of this analysis for some specific systems. As in the cases of component and initiating event estimates, it is possible to express Indian Point's analysis in terms of effective data and a conventional statistical analysis and thus assess the impact of their prior distributions and analysis methodology on their system results. Here we consider a general point.

In Section 0.16, the IPPSS authors make the excellent point (couched in Bayesian terms) that if a system contains two or more components whose failure probabilities are estimated by the same data, then this fact must be accounted for in estimating the system failure probability. Thus, for example, for two identical components in series for which the posterior mean and variance are α and β^2, respectively, the system failure probability has a posterior mean and variance of 2α and $4\beta^2$. If the two estimates were incorrectly assumed to be independent, the derived variance would be $2\beta^2$, which is too small. For two parallel components, the failure probability is p^2, say, which has a mean value of $\alpha^2 + \beta^2$. This is correct, but as a point estimate of p^2, this mean value can be very conservative.

Suppose one begins with a noninformative prior and modifies it with data, x/n, so that the posterior distribution has a mean of $p^* = x/n$ and variance = $p^*(1-p)/n$. Then, the posterior mean which is the Indian Point estimate of p^2 is:

$$\alpha^2 + \beta^2 = p^{*2} + p^*(1-p^*)/n$$

The expected value of this estimate (with respect to the sampling distribution of p^*) is (approximately):

$$E(\alpha^2 + \beta^2) = p^2 + 2p(1-p)/n$$

This result shows that, unless $(1-p)/n$ is much less than p, the Indian Point posterior mean value, regarded as an estimator of p^2, could be seriously biased (but in a conservative direction). This problem affects Indian Point's estimate of the probability of an interfacing system LOCA, which is one of their dominating contributors to risk.

From a Bayesian viewpoint, one could argue that both p and p^2 should not be estimated by their posterior means. In full-blown Bayesian analyses, a point estimate is selected on the basis of a loss function. If squared error loss is chosen (which means the penalty for estimating p by p^* is $(p-p^*)^2$), the posterior mean is the resulting estimator. However, squared error for p is not equivalent to squared error for p^2, so a Bayesian indiscretion occurs. Straightening this out is beyond the scope of this review. Section 0.16 of the IPPSS creates the impression that if one has selected a point estimate, say p^*, of p, with or without encumbering that estimate with lognormal connotations, then p^{*2} is unacceptable as a point estimate of p^2. Not so, by either Bayesian or statistical arguments.

2.6.8 Completeness

Another concern in risk estimation is completeness. What about accident sequences not covered in the report? In Section 0.19 of the Indian Point report, the authors discuss completeness. They argue that all possible initiating events are included in their list, that all possible resulting plant damage states have been identified, that the requisite system failures that lead to a damage state, given an initiating event are known, and that the combinations of component failures that fail a system are known. Thus, there is no set of damage-causing circumstances omitted from the study. (Note: This assumes that the fault and event trees are correct.) The only thing conceivably incomplete is the set of causes by which multiple component failures might occur. But, because the authors of Indian Point can put a number on this, everything is covered.

As an example, consider a system consisting of two identical trains. It can fail if (a) there are two independent train failures, (b) one train is out of service for maintenance and the other fails, or (c) one train has been disabled due to a human error and the other fails. Additionally, there may be (d) a human error or errors that disable one or both trains and there may be (e) support system failures that disable one or both trains. Indian Point considers all of these by conditioning on the state of a support system, generally electric power for which eight states are defined, then estimating the conditional probability of (a) through (d). Even so, it is recognized that there may be "other" causes of joint failure of the two trains. For example, there may be human or physical links not explicitly recognized. Indian Point estimates system failure probabilities for these situations in a variety of ways:

1. Inclusion of a ß-factor (ß).
2. Linkage to another estimate (L).
3. Judgment leading to a conclusion of negligible (N).

Table 2.6-4 shows the treatment of "other" failures in the IPPSS.

The ß-factor is in effect a factor to account for possible dependence between failure events. In the above example, if q denotes the failure probability of one train, then inclusion of a ß-factor leads to system failure probability of $q^2 + \beta q$, ignoring other terms in the system failure model. If we write this as $q(q + \beta)$, then $q + \beta$ corresponds to the conditional failure probability of the second train given failure of the first. In principle, ß can be estimated from

Table 2.6-4

IPPSS Treatment of "Other" Failures

System	IP-2	IP-3
Electric Power	L	L
Reactor Protection	L	L
Safeguards Act.	L	L
High Pressure Injection	β	β
Low Pressure Injection	β	N
Recirculation	β	β
Containment Spray	N	N
Fan Cooling	N	N
Component Cooling	N	N
Service Water	N	N
Auxiliary Feedwater	N	N

L = Linkage
β = β-factor
N = Negligible

data, but it is not in the IPPSS. Indian Point specifies their personal probability distribution for β as a lognormal distribution with a mean of .014 and a variance of 6.1(-4), which corresponds to 5th and 95th percentiles of .001 and .05. This "state-of-knowledge" is the same everywhere it is used.

The basis for Indian Point's assumed personal probability distribution for the β-factor is vague. A typical statement is the following:

> "Most of the observed coupled failures in the industry involved motor- or air-operated valves that had to change position on demand. The frequent partial tests and full refueling system tests indicate that an unforeseen common cause failure is of low frequency. This state of knowledge is expressed by taking a β-factor with range of 1.0×10^{-3} to 5.0×10^{-2} which yields a mean and variance of:
>
> $$\alpha_\beta = 1.4 \times 10^{-2}$$
>
> $$\beta_\beta^2 = 6.1 \times 10^{-4}$$" (p.1.5-483).

It would have been more straightforward for the authors to say, "We will model explicitly those dependencies we are aware of and deem important, such as by conditioning on electric power, and omit any others, because we feel they have negligible probability."

The one case in which the IP-2 and IP-3 analyses differed (low pressure injection system) is probably an oversight. Exactly the same words were used to discuss "other" failures. In only one case, though, were they followed by a β-factor calculation.

For the electric power systems, it was argued that "other" failures must be less likely than any specific failures, so the probability distribution assumed for the probability of "other" failures had its 95th percentile set equal to the smallest mean from an identified cause. For the other two systems where linkage was used, it was assumed that common calibration errors had the same probability as hardware failures. All of these "other" failures estimated by linkage had a negligible effect.

2.7 External Events

2.7.1 Seismic

In this section, the seismic external event is reviewed. The material in Sections 2.7.1.2 to 2.7.1.7 is based on a draft report prepared by Jack R. Benjamin and Associates, Inc. (JBA). Their report is contained in the Appendix B of this letter report. Appended to the JBA report are reports by Professors Ronald L. Street and Erik H. Vanmarcke which discuss the seismological aspects and the seismic hazard analysis. In addition, References (1-4) are referred to in the discussion of Section 2.7.1, and the comments given in References (5-7) were considered in the review.

The comments given in Sections 2.7.1.1 through 2.7.1.7 represent the most significant issues in the review and summarize the final conclusions. More detailed discussions of the issues can be found in the JBA draft report.

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6. Memorandum for Robert E. Jackson, from Leon Reiter, "Indian Point Probabilistic Risk Assessment (External Events)," May 14, 1982.

7. Memorandum for Edmund J. Sullivan, Jr., from Zoltan R. Rosztoczy, "Indian Point Probabilistic Safety Study-Seismic Equipment Fragility Review," May 19, 1982.

2.7.1.1 Seismic Logic Model

The seismic logic model is reviewed in Section 2.7.8.

2.7.1.2 Seismic Hazard

The methodology used in the IPPSS is appropriate and adequate to perform a seismic risk analysis. The procedure is well established and accepted. An important element of the seismicity studies conducted for the Indian Point site is the explicit treatment of the sources of variability in the analysis. The uncertainty in the analysis can be attributed to the limited data available on eastern U.S. seismicity and ground motion. This uncertainty is reflected in the final family of seismicity curves.

The two seismicity studies performed for the IPPSS by Dames and Moore (D&M) and Woodward Clyde Consultants (WCC) clearly identify the fact that variability due to modeling assumptions, or uncertainty as defined in the seismic fragility analysis, can contribute significantly to the variability in the fre-quency of exceedance curves. In addition, the statistical variability due to limited data and the inherent randomness of the process, which is combined with the modeling uncertainty, is also a significant contributor to the variablility in the final family of seismicity curves.

In generating the family of seismicity curves, the results of the D&M study have been modified in two ways. First, sustained-base peak acceleration values have been shifted by a factor of 1.23 to provide sustained acceleration; and second, the hazard curves have been truncated to reflect the belief that there is a maximum ground shaking intensity which can occur.

We believe that even if the curves had not been shifted there would be only a small change to the frequency of core melt analysis for Unit 2 and a moderate change for Unit 3. In general, we believe that a shifting factor F equal to 1.25 (which is essentially the same as the value of 1.23 used in the D&M report) is on the conservative side for structures. For equipment located in structures, which have a capacity below the capacity of the equipment, this value of F is probably also conservative.

Equipment, which does not have inelastic energy-absorption capacity or which depends on function capacity, respond more closely to the peak ground acceleration capacity. One example of this type of equipment is the service water pumps which depend on binding of the pump shaft for capacity and which are located at the ground level. However, the capacity of this component is relatively high and eliminating the 1.25 acceleration factor would not significantly change the results of the analysis.

We have adopted the following scale to quantify our comments in reviewing the IPPSS report:

Comment	Effect on Mean Frequency of Consequences or Core Melt
Small	Factor ≤ 2
Moderate	$2 <$ Factor ≤ 10
Large	Factor > 10

We agree that the upper-bound acceleration values applied to the D&M seismicity curves are reasonable. The WCC seismicity results were not modified in the main report as truncations were applied in the original study which is documented in the WCC section.

We believe that the truncation of the hazard curves should more appropriately have been performed within the probabilistic analysis. However, as verified by calculation, truncating outside the hazard analysis is conservative in that the annual exceedance frequencies for accelerations below a truncation level will be higher than had the truncation been performed in the probabilistic analysis.

In both seismicity studies, a Ramapo fault zone was not explicitly considered. However, in recent years considerable scientific study of the geology, historic and recent seismicity, have lead to a belief that a Ramapo fault zone is an alternative hypothesis that should be considered in the hazard analysis (Ref. 1, 2, 3, and 4). Since, the geometry of the fault zone, seismicity parameters, and a maximum event size are difficult to determine we feel that a family of seismicity curves for a Ramapo fault should be considered. The absence of the Ramapo zone from the final family of seismicity curves is, in our judgment, an inadequacy in the analysis.

We agree that the overall seismic hazard methodology utilized by D&M and WCC is appropriate and adequate to determine frequency of exceedance curves on levels of ground shaking. Although the general probabilistic methodology is the same in both studies, there are differences in how the ground motion models were applied, the selection of key parameters, and the definition of seismic source zones. In our judgment, the WCC study does not accurately represent the uncertainty in the earthquake process. Because of the low upper-bound intensity values used (i.e., VII and VIII) in the WCC study; we believe that the seismic hazard is better represented by the D&M study.

2.7.1.3 Seismic Fragility

The methodology used in the IPPSS report for determining seismic fragility effects is appropriate and adequate to obtain a rational measure of the probability distribution of the frequency of core melt and associated release categories.

Structural failure is defined as ". . . The onset of significant structural damage, not necessarily corresponding to structure collapse." This definition may be conservative in some cases and will tend to produce higher frequency of failure estimates compared to a definition based on collapse where functional failure is not an issue. It would be more appropriate to use a median definition and add uncertainty for the definition. We agree that it is appropriate to define failure as either rupture/collapse or loss of function, whichever occurs first.

We agree with separating variability of seismic response and structural capacity into randomness and uncertainity components.

Use of the lognormal distribution is appropriate as long as the extreme tails of the density function do not significantly influence the results of the analysis. It was found in performing the integration of the hazard and fragility curves that most of the contribution (i.e., greater than 90 percent) to the release category 2RW for Indian Point 2 was within three standard deviations from the median value for the control building/superheater building impact fragility distribution which controlled the system fragility curve for 2RW. In contrast, the contribution to release category 2RW for Indian Point 3 was generally beyond three standard deviations from the effective median value of the structure components which contribute to the mean frequency value of 2RW (i.e., the control building and diesel generator fuel oil tanks at approximately 0.8g). We believe that the results for Indian Point 3 using the lognormal distribution are conservative since the lower tail of the lognormal density function tends to be higher than other reasonable distributions which could have been used. However, as stated in Chapter 2, neglecting possible design and construction errors may overcompensate the possible conservatism in using the lognormal distribution.

After reviewing the procedures used to produce the fragility data, we have a general impression which bears on the issue of consistency. We feel that the uncertainty of the parameters in the IPPSS report has probably been understated. There are various levels of sophistication which have been used to develop the fragility parameter values, but we do not sense that enough uncertainty has been assigned to components where parameter values are based on more distant information. Although in fairness to the IPPSS report, the values for βu are generally larger for generic components as compared to plant specific components. On the other hand, we also believe that the median capacity values are probably low.

Several obvious elements of uncertainty have been left out of the seismic fragility analysis. First, design and construction errors (e.g., the problem of piping supports at Diablo Canyon) and aging effects are not included in the seismic fragility or fault tree analysis. These become extremely important issues for series systems such as piping and cables (i.e., cable trays). One failure and the system may be lost. We noted for several sections which we reviewed that the authors did not check the calculations which formed the basis for the fragility parameters that were developed. Thus, errors in the calculations could not be discovered.

One approach used to develop fragility curves was based on analysis of generic data. Rather than working with the analysis of a plant specific component, failure and/or response data from similar components in similar environments are used as the basis to develop a fragility curve for the particular plant component being considered. We feel this procedure is appropriate under certain circumstances. If after determining the fragility of a particular plant component using generic data it is found that the capacity is sufficiently high so that the component does not influence the release category analysis, then we feel the analysis is appropriate. On the other hand, if the component is found to have a low capacity such that it influences (or could if changed by a small amount) the frequency of core melt analysis, then a more detailed analysis for that component should be conducted.

As a result of our tour of the Indian Point Site, we question whether the IPPSS has considered all possible failures of nonsafety-related structures or equipment, which could impact on safety-related items. The IPPSS has included, for example, possible failure of the stack, superheater building, and the turbine building onto the Unit 2 control building for seismic loads. It was pointed out during the tour that the nitrogen bottles in the Unit 3 AFW pump room could fail and the released gas propel them into safety-related control cabinets.

This type of secondary failure was not considered in the analysis. Another possibility which was not documented in the IPPSS report is potential failure of the polar crane structures in the containment buildings and possible failure onto equipment below. We believe that a systematic study should be conducted to identify and quantify the effects of possible secondary failures which could affect safety-related structures and equipment.

2.7.1.4 Sensitivity Analysis

In order to understand how changes in the analysis parameters might affect the mean frequency of release category 2RW, we performed a sensitivity analysis using the same discrete probability distribution procedure used in the IPPSS report. The mean frequency values given in the report for 2RW are 1.4 x 10-4 per year for Unit 2 and 2.4 x 10-6 per year for Unit 3, which were used for comparison.

The hazard curves from IPPSS report Sections 7.9.1 and 7.9.2 were used in the sensitivity analysis. The relative weights which were assigned were the same as used by IPPSS. The fragility curve values for release category 2RW were obtained from Table 7.2-4 for Unit 2 and Table 7.2-8 for Unit 3 from the IPPSS report.

The purpose of the sensitivity study was to determine the differences between the D&M and the WCC seismicity curves and to investigate the effects of shifting and truncating the curves. The D&M curves were shifted by a factor of 1.23 (this was done to convert from peak ground acceleration to damage-effective ground acceleration) and truncated for assumed upper-bound cutoff values (see discussion for IPPSS report Sections 7.2 and 7.9.4). The WCC curves developed in Section 7.9.2 were based on a damage-effective ground acceleration parameter and were also similarly shifted and truncated. (See discussion for IPPSS report Sections 7.2 and 7.9.2)

The results of the sensitivity analysis are presented in Table 2.7.1-1. The combined results for the shifted and truncated curves at the bottom (0.8 x 10-4 for Unit 2 and 1.6 x 10-6 for Unit 3) should be the same as the IPPSS results for Units 2 and 3. We believe that the difference is due to the procedures used to perform the integration and the coarseness of the hazard and fragility data points. In addition, there may be some difference due to the lumping of curves done in the IPPSS analysis (Figure 7.2-4 does not replicate the seven D&M curves from Figures 7.2-1 and 7.2-2 and the four WCC curves from Figure 7.2-3). In some sense, the difference in the results represent an analysis procedure error or uncertainty.

TABLE 2.7.1-1
RESULTS OF SEISMIC SENSITIVITY ANALYSIS

	Mean Frequency, Release Category 2RW (per year)	
Category	Unit 2	Unit 3
D&M		
Unshifted and Untruncated	2.6 x 10-4	1.1 x 10-5
Shifted and Truncated	1.5 x 10-4	3.2 x 10-6
WCC		
Unshifted and Untruncated	1.7 x 10-4	1.6 x 10-6
Shifted and Truncated	1.3 x 10-5	3.5 x 10-9
Combined Results		
Unshifted and Untruncated	2.2 x 10-4	6.2 x 10-6
Shifted and Truncated	0.8 x 10-4	1.6 x 10-6
IPPSS Results	1.4 x 10-4	2.4 x 10-6

In general, we believe that the data points for the hazard and fragility curves in the IPPSS are too coarse. A more refined set of points should be developed.

Several conclusions can be made based on the results of the sensitivity analysis.

1) The mean frequency of release category 2RW for Unit 2 is greater by a factor of approximately 12 between the D&M and the WCC <u>shifted and truncated hazard curves</u> (i.e., 1.5 x 10-4 per year compared to 1.3 x 10-5 per year). These are the curves ultimately used in the IPPSS analysis. Note that for Unit 3 the difference is about a factor of 1000. The reasonableness of this result is discussed for IPPSS report Section 7.2 in the JBA report (see Appendix). Based on this study, it is clear that the WCC hazard curves are considerably different from the D&M curves.

2) For the D&M hazard curves the difference between the unshifted and untruncated results and the modified results is a factor of less than 2 for Unit 2 and slightly over 3 for Unit 3. The low factor for Unit 2 is because the median fragility value of 0.27g for Unit 2 is well away from the upper-bound cutoff values. For Unit 3 the effective median fragility value of 0.8g, is at the upper limit of the cutoff values. Note that plots of the hazard curves are given in IPPSS Figures 7.2-1 through 7.2-4.

3) For the WCC hazard curves, the difference between the unshifted and untruncated results and the modified results is a factor of 13 for Unit 2 and a factor of almost 500 for Unit 3. The high factors for both units is because the median fragility values are at or above the upper-bound cutoff values.

4) The difference between the shifted and truncated combined results (which are the basis for the final values given in the IPPSS report) for Units 2 and 3 is over two orders of magnitude. The reason is due to the effective capacity for Unit 2 being 0.27g and for Unit 3 being 0.8g.

2.7.1.5 Ramapo Zone Investigation

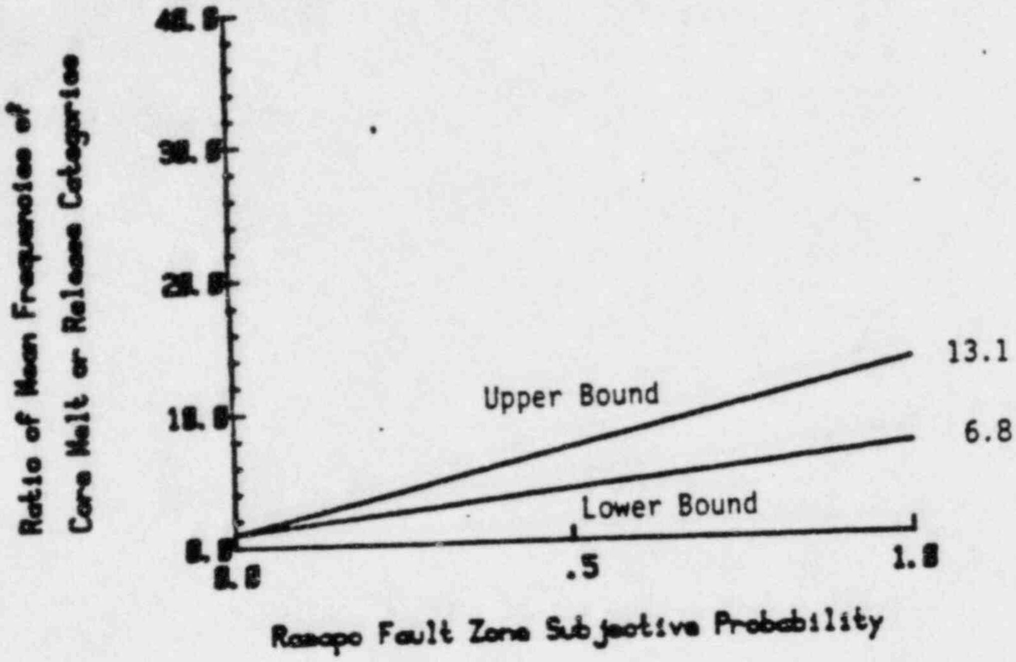
The increase in the mean frequency of release category 2RW due to different representations of a Ramapo fault zone were calculated using a seismic hazard model. The results show an increase due to the Ramapo source in comparison to mean frequency values obtained in the IPPSS. We postulated, in a Bayesian sense, a subjective weight for the Ramapo source and then combined this source with the other postulated sources. Based on the information we have to date, we are unable to make a formal assignment for the Ramapo source. However, we have investigated the implication of various weights which could be assigned. At one limit is the probability assigment of 0. This implies that the Ramapo source is incapable and thus cannot possibly occur. At the other extreme is the probability assignment of 1.0 which says that the Ramapo source, plus a reasonable background seimicity which was added, replaces the other source zones considered in the IPPSS. This is obviously a very conservative scenario since it is highly unlikely that the only possibility is the Ramapo zone. For purposes of this sensitivity analysis, the D&M Piedmont zone with a M5.7 maximum magnitude is selected to be the background seismicity. This is also conservative.

Because there is a difference in integration procedures used by IPPSS and us, we have normalized the increase in mean frequency of consequences to correspond to the values given in the IPPSS report. In this investigation we have not included any other differences which we found in our review. Thus, the results presented here are given in addition to changes we noted elsewhere in this section.

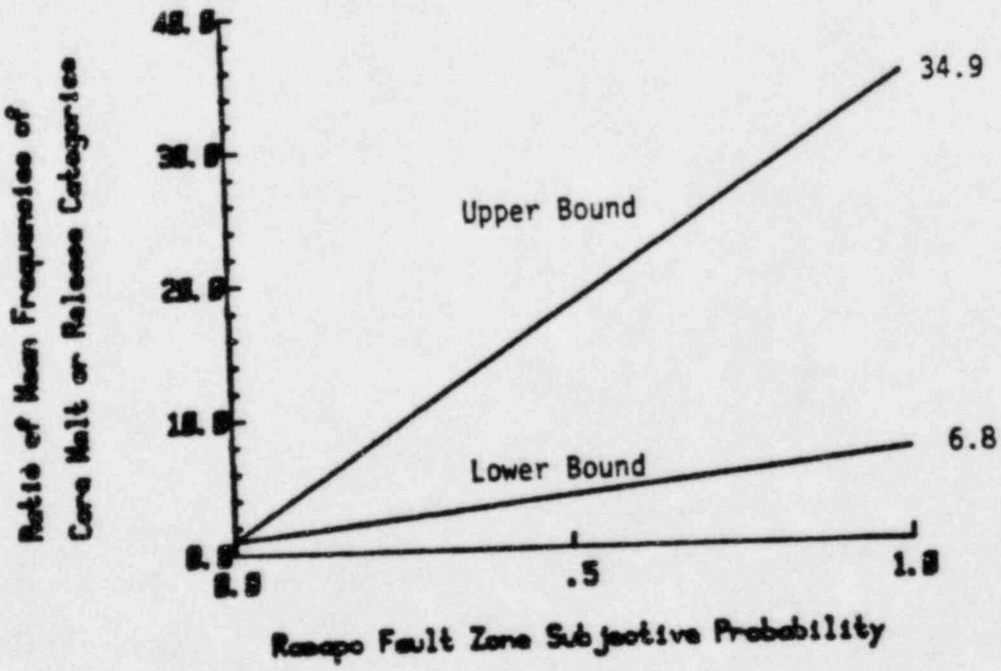
Figure 2.7.1-1 shows the effect of the Ramapo fault zone and its assumed background seismicity on the mean frequency of core melt or release categories for subjective probability values between 0 and 1. The curves were developed for release category 2RW. However, we expect the trend to be similar for other release categories and for core melt as well. Curves given for both Unit 2 and Unit 3 represent the <u>ratio</u> of the total seismicity-caused mean frequency (including the weighted contribution from the Ramapo source and background seismicity) to the seismically-caused mean frequency values corresponding to the IPPSS report (i.e., 1.4×10^{-4} per year for Unit 2 and 2.4 x 10-6 per year for Unit 3). Thus the results shown in Figure 2.7.1-1 pertain only to seismically-caused consequences. The two curves shown for each plant represent lower and upper bound possible Ramapo fault zones.

Figures 2.7.1-2 and 2.7.1-3 show similar plots for total mean frequency of release category 2RW and core melt, respectively. In these plots the mean frequency values given in IPPSS report Tables 8.3-2 and 8.3-3 were used as the base values for Unit 2 and Unit 3, repectively. Thus the effect of the Ramapo fault zone on higher level consequences as function of the subjective probability for the zone can be seen.

By comparing Figures 2.7.1-1 through 2.7.1.3, it is seen that the effect of the Ramapo fault zone decreases monotonically from seismic-caused release categories, to total release category 2RW, and finally to core melt. The reason the



(a) Unit 2



(b) Unit 3

Figure 2.7.1-1 Effect of Including a Ramapo Fault Zone on Seismic-Caused Consequences

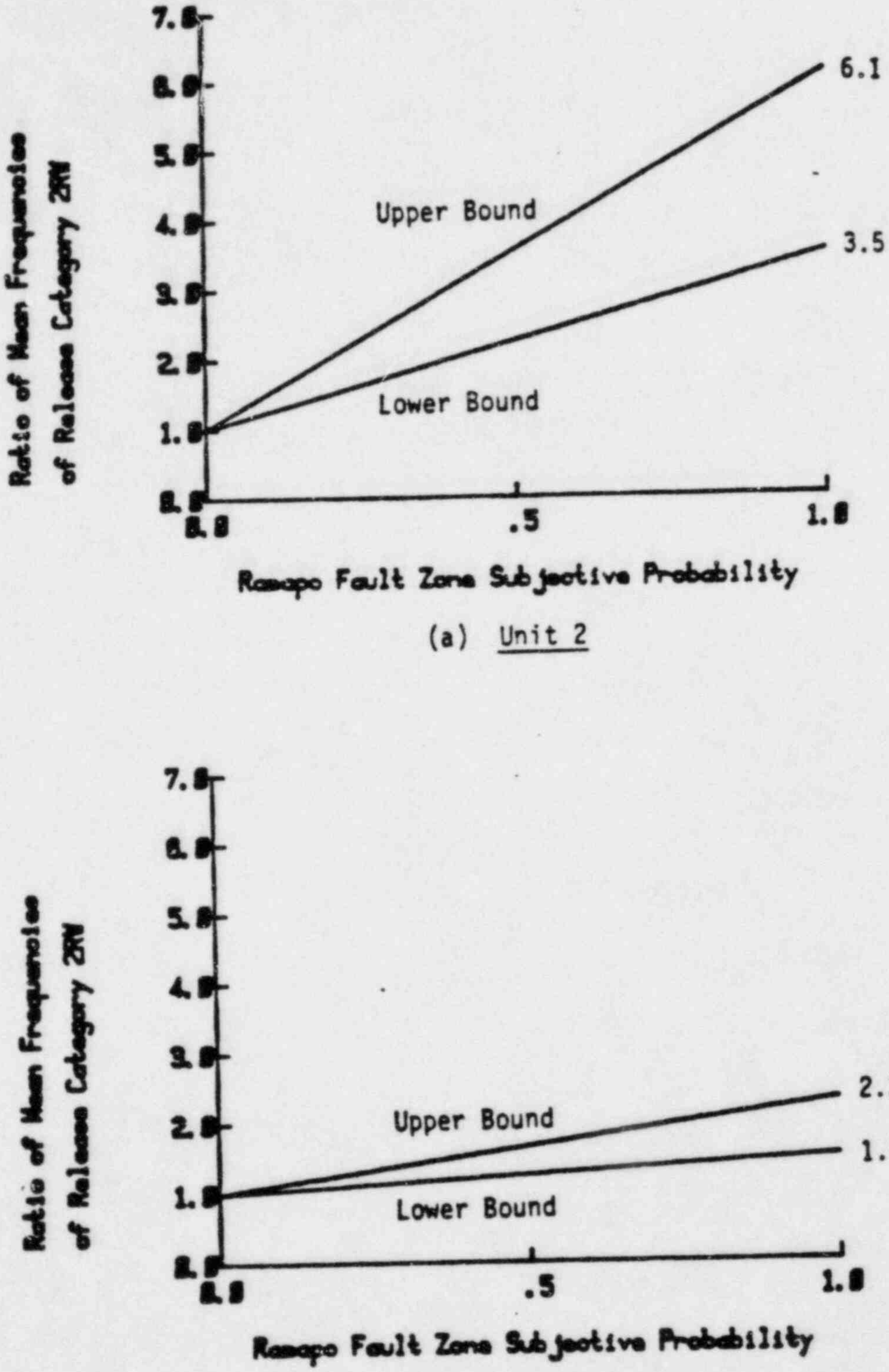
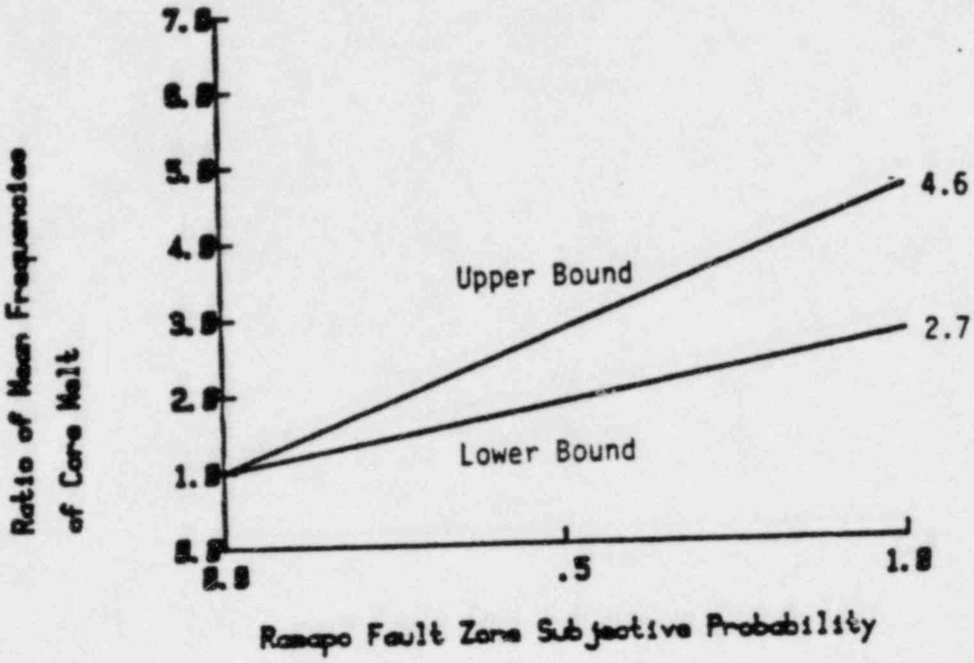
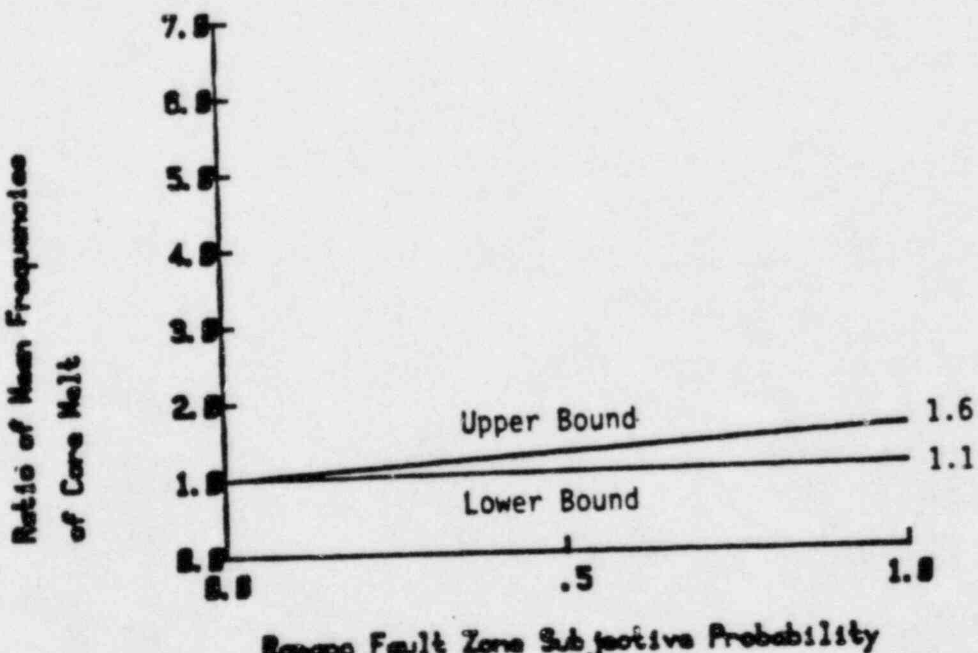


Figure 2.7.1-2 Effect of Including a Ramapo Fault Zone on Total Release Category 2RW



(a) Unit 2



(b) Unit 3

Figure 2.7.1-3 Effect of Including a Ramapo Fault Zone on Total Core Melt effect of the Ramapo decreases is because other events such as fire, hurricane, tornado, and internal accidents dilute the contribution made by the Ramapo source.

2.7.1.6 Systems Analysis

We believe that the mean value of 1.4 x 10-4 per year for the annual frequency of core melt for Unit 2 is low by a factor of 2 because of the hazard curves. We feel that the hazard frequency of exceedance values are better represented by the values given by D&M. Since the D&M and WCC seismicity curves were weighted equally, and since the mean frequencies of core melt based on the WCC curves are more than a factor of 10 lower, (see sensitivity investigation above), the IPPSS values are doubled if only the D&M hazard curves are used. As discussed in the review of Section 7.9.3 in the JBA report (see Appendix), we believe that the fragility for the dominant component (i.e., impact between Units 1 and 2 control rooms) is conservative. A more detailed analysis of this failure mode would probably lead to a higher median capacity.

Because of the higher level of subjective uncertainty leading to the tails of the core melt frequency density function, we do not believe the reported 90 percent confidence bounds for Unit 2 are credible.

Because component 2 (impact between control rooms of Units 1 and 2) dominates the analysis, possible dependence between capacities and/or responses of other components does not affect the analysis results for Unit 2.

In regards to the systems analysis for Unit 3, the control building median capacity is given as to 1.20g, which is based on a shear wall failure mode. We believe that this value may be high (i.e., unconservative) for the Unit 3 control building. Part of the argument for developing the capacity for the Unit 2 control building was that impact between Unit 1 and Unit 2 control rooms will cause failure of the hung ceiling which would fall and incapacitate all operators. Based upon visual inspection of both Unit 2 and 3 control rooms, we found that both ceilings are hung by wires without special seismic bracing. We doubt that the control room ceiling for Unit 3 has a capacity equal to 1.20g damage-effective ground acceleration. The dominant components for core melt and release category 2RW are the control building shear wall and the diesel generator fuel oil tanks which together have an equivalent median capactiy of about 0.8g. We doubt that the hung ceiling in the control room has a capacity even that high.

The capacity of the diesel generator fuel oil tanks, which are buried, are based on generic data. Because this component contributes significantly to core melt and 2RW for Unit 3, a specific analysis for this component should be conducted.

It is doubtful that any dependence between the components will affect the analysis results for Unit 3. Note that perfect dependence due to ground motion is implicitly assumed in the procedure for integrating the hazard and fragility curves. Since the control building and fuel tanks are separate structures, no capacity or other response dependence is present.

We believe that for the mean value of 2.4 x 10-6 per year for the annual frequency of core melt for Unit 3 may be low due to potential failure of the control room ceiling and our belief that the D&M hazard curves are more representative of the Indian Point site. We feel that these differences would change the reported value by a factor of about 8. We do not believe that the reported 90 percent confidence bounds are credible.

2.7.1.7 Conclusions and Recommendations

We believe that certain results may be unconservative. Based on our review Table 2.7.1-2 gives a revised list of mean frequencies for Indian Point Unit 2. Table 2.7.1-3 gives a similar list for Unit 3. Below each of the mean frequencies for seismic, hurricane, and tornado is the ratio of the revised value to the value given in the IPPSS report (see Tables 8.3-2 and 8.3-3 for the IPPSS report values for Units 2 and 3, respectively).

We believe that the D&M hazard curve values are more representative of the Indian Point site; thus, we choose to weigh the D&M curves with a probability value equal to 1.0. This assumes that the results for a Ramapo source are contained within the family of seismicity curves developed by D&M. Because the release category frequency values for the WCC curves are about an order of magnitude below the frequency values based on using the D&M curves, the mean release category and core melt frequencies for Unit 2 are approximately doubled. (Note that the D&M and the WCC values were each weighted equally in the IPPSS.)

We believe that for Unit 3 the capacity of the hung ceiling in the control room may be lower than the equivalent median capacity value of 0.8g, implicitly used in the IPPSS. We estimate that the mean frequency for release category 2RW, which has a dominant contribution from the control building, increases by a factor of 5. Similar to the revised values for Unit 2 for the increase in the hazard function, we increase the

TABLE 2.7.1-2

REVISED MEAN RELEASE FREQUENCIES - UNIT 2

Release Category	Seismic	Hurricane	Tornado
Z-1Q	1.4×10^{-6}** (2)***	0	0
Z-1	2.6×10^{-8} (2)	small	small
2	5.8×10^{-8} (2)	small	small
2RW	2.8×10^{-4} (2)	5.4×10^{-4} (20)	1.6×10^{-5} (1)
8A	8.4×10^{-9} (2)	0	small
8B	5.2×10^{-10} (2)	0	0
Core Melt	2.8×10^{-4} (2)	5.4×10^{-4} (20)	1.6×10^{-5} (1)

**Mean Frequency
***Ratio to IPPSS Value

TABLE 2.7.1-3
REVISED MEAN RELEASE FREQUENCIES UNIT 3

Release Category	Seismic	Hurricane	Tornado
Z-1Q	7.4×10^{-8}** (2)***	0	0
Z-1	5.0×10^{-9} (2)	0	small
2	small	0	small
2RW	2.4×10^{-5} (10)	0	9.2×10^{-7} (1)
8A	1.4×10^{-6} (2)	0	4.1×10^{-7} (1)
8B	4.4×10^{-7} (2)	0	0
Core Melt	2.6×10^{-5} (7.9)	0	1.3×10^{-6} (1)

**Mean Frequency
***Ratio to IPPSS Value

mean frequencies of all categories by an additional factor of 2 to produce a total factor equal to 10 for release category 2RW and a factor of 2 for other categories. Core melt due to seismic increases by a factor of almost 8.

In order to resolve the most significant issues which have been raised in the review, we recommend the following be done.

1) For Unit 3, the capacity of the hung ceiling in the control room should be analyzed and a fragility curve developed for this component and incorporated into the plant analysis.

2) For Unit 3, the capacity for the diesel generator fuel oil tank, which is a dominant contributor, should be based on a specific analysis for this component. Generic-based values were used in the IPPSS.

3) The Ramapo Fault should be represented in the seismic hazard analysis (i.e., area, recurrence distribution, upper-bound magnitude, etc.) and probability weight(s) assigned.

2.7.2 Wind

In this section, the wind external event is reviewed. The material in Sections 2.7.2.2 to 2.7.2.6 is based on a draft report prepared by Jack R. Benjamin and Associates, Inc. (JBA). Their report is contained in Appendix B of this letter report. Appended to the JBA report is a report by Dr. Larry A. Russell who discusses the hurricane hazard analysis. In addition, References (1-5) are referred to in the discussion of Section 2.7.2, and the comments given in References (6-7) were considered in the review.

The comments given in Sections 2.7.2.1 through 2.7.2.6 represent the most significant issues in the review and summary of the final conclusions. More detailed discussions of the issues can be found in the JBA draft report.

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6. Memorandum for Edmund J. Sullivan, Jr., from Franz P. Schauer, "Indian Point Probabilistic Safety Study--Review Comments on Section 7.5 (Structural Fragility)," undated.

7. Memorandum for Edmund J. Sullivan, from Earl Markee, "Indian Point Probabilistic Safety Study-External Events," May 11, 1982.

2.7.2.1 Wind Logic Model

The wind logic model is reviewed in Section 2.7.8.

2.7.2.2 Wind Events

We concur with the procedure to develop hazard curves for extreme winds, hurricanes, and tornadoes separately, and the assumption that the results from the three sources are independent. We believe that correction factors for the effects of height, which were included in the analysis, are small relative to the influence of adjacent structures, which were not explicitly included in the analysis.

We believe that the tornado hazard curves are conservative, but that the hurricane hazard curves are unconservative. The implications of this result are discussed below.

2.7.2.3 Tornado Missiles and Winds on Concrete Structures

The statement that the concrete stuctures were designed for 25 psf wind loading, and that there is "little deflection" is misleading and not pertinent to the conclusion that potential wind pressures and tornado missiles are not significant to Indian Point safety-related <u>concrete</u> structures (i.e., wall thickness greater than 12 inches). We concur with this conclusion based on review of References 1, 2, and 3.

The statement that tornado frequencies at Indian Point are lower should be documented (although we do agree with this statement). In general, other leading statements made in this section should be documented.

2.7.2.4 Tornado Missiles and Winds on Metal Structures

We agree that it is conservative to base the fragility of metal structures and exposed equipment on the hit frequency; however, the fragility curves for the effects of tornado missiles were not developed based on possible hit frequencies as stated, but rather on wind velocities which could lift various missiles off the ground. However, we believe that using the tornado impact fragility curves shown in IPPSS report Figure 7.5-3 results in conservative frequencies of failure for the structures and equipment considered. We developed our basis for this conclusion using References 2 and 3 which reported the probability of hit frequency of specific structures at a nuclear power plant. The basis for our opinion is documented in the JBA report (see Appendix B).

We feel that hurricane-caused missiles are probably not a problem; however, this potential cause of failure should be considered and documented in the IPPSS report.

We believe that the major uncertainty in wind loading on an Indian Point structure (conditional on the occurrence of free-field wind velocity) is due primarily to the influence of nearby structures. We do not believe that the randomness or uncertainty included for the capacity due to wind have been rationally developed in the IPPSS to include the influence of the close proximity of adjacent Indian Point structures. Also, we disagree with the development of the wind load correction factor SF_L.

For hurricane winds, SF_L randomness was based on consideration of differences in terrain and return period occurrence wind speeds. The influence of nearby structures is more significant than terrain variability and should have been explicitly included. Also differences in occurrence rate belongs in the wind speed hazard analysis rather than the fragility formulation. For tornadoes, SF_L, randomness was based on the relatively insignificant differences in wind speed effects over the height of the structures.

Because of the approach used to develop the factor SF_L, the slope of the fragility curves for tornado effects are steep while the corresponding curves for hurricanes are less steep. We believe that the randomness (which is expressed by the slope of the fragility curves) should be essentially the same for the effects of tornado and hurricane wind speeds.

We noted two discrepancies in the development of the fragility curves. In Table 7.5-1, the velocity pressure for exposure C for a 100-year return period from Reference 4 should have been 27 psf instead of the value of 18.5 psf used in the analysis. The effect of this error would be to increase the randomness for hurricane wind fragility curves which would lead to a slightly larger frequency of core melt (probably a small effect).

The second discrepancy is the conversion of pressure to equivalent wind velocity using the equation: $q = 0.00256 V^2$ (where q = psf, V = mph). This equation ignores the differences between structure shapes. For example, a rectangular building in the open is more closely modeled by the equation of $q^* = 1.3q$ where 1.3 is the shape factor. Because of the influence of adjacent buildings, the shape factor will vary from structure to structure. We believe that the only rational way to develop shape factors for buildings at Indian Point is through use of a wind tunnel model. Our judgment is that the shape factors for the Unit 2 control building, the Unit 2 diesel generator building, and RWST also vary depending on the type of failure being considered. As discussed below, these structures control the core melt and release frequency analysis.

Assuming a local failure may control the capacity of the diesel generator building, the median capacity may be smaller by a factor as much of 1.7; however, this building is shielded to some extent. For the RWST we believe that the implicitly-assumed shape factor of 1.0 is appropriate. Because of the location of the control building, which is relatively sheltered, the shape factor is probably 1.0 or less. However, this should be confirmed by IPPSS personnel and documented.

The offsite power fragility is assumed in the IPPSS to be controlled by the fragility of the transmission line towers. Because the offsite towers have not been specifically identified and analyzed, we believe that a median fragility wind velocity value of 140 mph is unconservative. It is likely that offsite power will be lost at a much lower wind velocity. We believe that it would be prudent to assume that offsite power is not available if either a tornado or hurricane occurs. The implication of this assumption is discussed below.

We feel that there is no rational basis for the assumption that the upper-bound and lower-bound fragility curves are each weighted with probability 0.1. The result of this assignment causes the three middle fragility curves used for the hurricane and tornado analysis (see IPPSS report Tables 7.5-4, 7.5-5, and 7.5-6) to be nearly identical. Because of the apparently arbitrary assignment of probability values (i.e., 0.2 could have equally been used for the upperand lower-bound curves), we do not have confidence in the spread of the probability distribution. Also, the mean values will change significantly for hurricanes as the probability assignments are altered. This is due to the relative steepness of the hurricane hazard curves.

We believe that the possibility of either the turbine building or the superheater building failing and falling on the control building should be considered. Also the possibility of the superheater building failing and falling on the diesel generator building and the condensate storage tank should be considered. The fragility curves for these structures should be developed to determine whether they effect the probability of core melt and subsequent release.

2.7.2.5 Systems Analysis

Based on the fault trees given in IPPSS report for Unit 2, Figures 7.5-6 a through f, the Boolean equations leading to core melt, M_W were checked. We generally agree with the final expression given on page 7.5-12. We believe that part of the probability of the stack failing and falling on either the control building or the diesel generator building was omitted. This contribution amounts to $0.05\ 7_W\ V\ 0.05\ 7_T$. Because of the high capacity of the stack relative to the control and diesel generator buildings, this discrepancy has no significant impact.

The significant contributors to core melt for Unit 2 are due to wind pressure failure of offsite power 2_W, the control building 4_W, and the diesel generator building 6_W. Note that the subscript "W" refers to either hurricane or tornado winds, while "T" refers only to tornado missile effects. The significant portion of the core melt Boolean equation is $2_W \wedge (4_W\ V\ 6_W)$. The other parts of the equation are not important since the capacity for tornado missiles is relatively high.

We believe that the mean annual frequency of core melt value of 4.3 x 10-5 per year for Unit 2 may be low by a factor of about 13 (a large effect). We do not believe that the confidence bounds given are meaningful.

Based on the fault trees given in the IPPSS report, Figures 7.5-11 a through e, the Boolean equations leading to core melt, M_W, for Unit 3 were checked. We agreed with the equations given in the IPPSS report.

The significant contributions to core melt are due to failure of either the RWST, 9_T, or the service water pumps, 11_T. Other components in the sequence, such as offsite power and the AFW pump building, will fail due to wind pressure at much lower wind velocities than missile failure of the RWST or the service water pumps.

We believe that the mean annual frequency of core melt value of 1.3 x 10-6 per year for Unit 3 is reasonable. We do not feel that the confidence bounds given are meaningful.

2.7.2.6 Conclusions and Recommendations

We believe that certain results may be unconservative. Based on our review, Table 2.7.1-2 gives a revised list of mean frequencies for Indian Point Unit 2. Table 2.7.1-3 gives a similar list for Unit 3. Below each of the mean frequencies for seismic, hurricane, and tornado is the ratio of the revised value to the value given in the IPPSS report (see Tables 8.3-2 and 8.3-3 for the IPPSS report values for Units 2 and 3, respectively.)

Two factors produce an estimated increase in release category 2RW for Unit 2 due to hurricane effects. Based on review of Section 7.9.5, we believe that the median hurricane hazard curve is unconservative. A comparison of the IPPSS median and upperbound curves was made with hazard values obtained from Reference 5. Using a range of hazard curves based on this reference and the median fragility curve from IPPSS Table 7.5-4, we obtain a factor of 10 to 30 increase in release category 2RW. We believe that a factor of 10 increase is appropriate for differences due to the hazard curves. In addition, the hazard curves do not include complex site roughness boundary layer effects and wind channelization caused by local hills and the Hudson River Valley. Due to a higher estimated hazard curve, the frequency of 2RW and core melt are judged to increase by a factor of 10. Also because offsite power probably will be lost at a wind speed below 140 mph, the frequency of 2RW release and core melt increase by a factor of 2. The total factor for both these effects is a 20-fold increase in mean frequency for 2RW and core melt for Unit 2.

For tornado wind effects, we believe that the capacity of offsite power has been assumed too high. We estimate that the frequency of release category 2RW increases by a factor 2 for Unit 2. However, we judge that the hazard curves are conservative by at least an equivalent factor; thus, we believe that the IPPSS mean frequency values for 2RW and core melt are reasonable.

Hurricane winds are not a significant event for Unit 3.

Since the frequency of release depends on tornado missile impact, we judge the IPPSS results for tornado hazard to be reasonable for Unit 3.

In order to resolve the most significant issues which have been raised in the review, we recommend the following be done.

1) A fragility curve for offsite power should be developed which considers various possible failure mechanisms (i.e., in addition to the failure of the transmission towers).

2) Wind fragility curves should be rationally developed for the Unit 2 control building and the diesel generator building. They should explicitly consider the structure shapes and the effects of adjacent structures. Possible local failure

of siding and roofing should be considered in determining the structure capacities. Also, the fragility of the Unit 1 turbine and superheater buildings should be calculated for wind. The possibility of these buildings failing and falling on safety-related structures (i.e., Unit 2 control building, diesel generator building, and condensate storage tank) should be included in the plant analysis.

3) A hurricane hazard analysis which includes careful evaluation of the site roughness boundary layer effects and wind channelization by the local hills and river valley should be performed.

4) A systematic comparison between the hurricane hazard curves given in the IPPSS and the results in Reference 5 should be made to provide the basis for the large differences that exist and justification of the reasonableness of the IPPSS results.

2.7.3 Flooding

In this section external and internal flooding events are reviewed. The material in Sections 2.7.3.1 and 2.7.3.2 is based on a draft report prepared by Jack R. Benjamin and Associates, Inc. (JBA). Their report is contained in Appendix B of this letter report. References 1 through 8 are referred to in the discussion in Section 2.7.3.1 and References 9 through 14 are referred to in Section 2.7.3.2. The comments given in Reference 15 were considered in the review.

The comments given in Sections 2.7.3.1 and 2.7.3.2 present the most significant issues in the review and summarize the final conclusions. More detailed discussions of the issues can be found in the JBA draft report.

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14. Letter from Steven A. Varga, Chief, Operating Reactor Branch #1, Directorate of Licensing, U.S. Nuclear Regulatory Commission to Mr. John D. O'Toole, Assistant Vice President, Nuclear Affairs and Quality Assurance, Consolidated Edison Company of New York, Inc., dated December 18, 1980.

15. Memorandum for Ted Sullivan, from George Lear, "Indian Point-External Events," May 18, 1982.

2.7.3.1 External Flooding

The Indian Point plant is situated on the east bank of the Hudson River, approximately 43 miles north of New York City. The plant elevation is approximately 14.0 ft. which corresponds to the elevation of the screenwall structure for Unit 3 (Ref. 1). The plant grade is about 15 ft. The consideration of potential flooding at the site due to external flood is based principally on the flood studies conducted for the Indian Point Unit 3 operating license review, (Ref. 2, 3, 4, 5, and 6). The design basis of Unit 3 for external flooding, and thus the IPPSS, is based on the occurrence of extreme events and event combinations such as the Probable Maximum Flood (PMF), the Probable Maximum Hurricane (PMH), high tides, etc. The IPPSS concludes the contribution to the frequency of core melt due to external flood sources is extremely small. The basis of this conclusion is reviewed and the adequacy of the probabilistic methodology is discussed.

The principal basis of the external flooding analysis in this section is the work in Reference 4, and various supplements or revisions (Ref. 5, 6). The intent of these studies was to evaluate maximum water surface elevations at the site. On the basis of a review of potential sources of flooding on the Hudson River, the following events and event combinations were considered:

- Probable Maximum Precipitation (PMP), which is assumed to produce the Probable Maximum Flood (PMF)
- PMF and tidal flow
- Standard Project Flood (SPF) and Ashokan Dam Failure
- SPF and the Standard Project Hurricane (SPH) at New York Harbor
- SPF, Ashokan Dam Failure and the SPH at New York Harbor
- Probable Maximum Hurricane (PMH) and spring high tide.

The result of deterministic calculations for these events are provided in Reference 6. The IPPSS estimates of the annual frequencies of occurrence of individual events range from 10^{-3}/yr to 10^{-4}/yr, while frequencies of event combinations have estimated values of 10^{-8}/yr to 10^{-12}/yr. The IPPSS concludes on the basis of the foregoing results that the contribution of external flooding to the annual frequency of core melt is extremely small. For this reason the study does not consider the impact of flooding on safety-related equipment or structures.

The approach taken in the IPPSS to assess the frequency of external flooding at the Indian Point site is to consider only the most extreme events (i.e. probable maximum events), and event combinations. The reason for this is apparently the fact these events were the basis of the flood design criteria used for the Indian Point site. This approach differs from a probabilistic flood hazard analysis that considers a full complement of water elevations due to a range of event sizes. The IPPSS has in effect chosen to consider for a given source of flooding one or two events and their resultant water surface elevations produced at Indian Point.

The approach taken to evaluate the chances that external flooding would affect safety-related equipment does not appear adequate. We feel that the methodology employed has not adequately treated the sources of uncertainty in the analysis which may be large. Relevant examples of the uncertainty in flood routing and water surface elevation mapping, including the uncertainty in flood routing procedures, are presented in Reference 7. The study suggests that an average value of the one standard deviation in the estimate of water surface profiles due to riverine flooding is approximately 23 percent of the estimated flood depth. In addition, other sources of uncertainty include the frequency of occurrence model employed, the uncertainty about a derived frequency function, storm selection, etc.

At a meeting with IPPSS personnel our concern that the uncertainty in the flood analysis was not taken into account was expressed. The response by IPPSS provided in Reference 8, does not address this issue.

It is not apparent in the analysis for flooding due to hurricanes that an occurrence at a location other than New York Harbor is considered. This approach is not consistent with the probabilistic hurricane analysis in Section 7.9.5.

We conclude from our review that the sources of external flooding at the Indian Point site have been identified and adequately considered in a <u>deterministic</u> sense.

The probabilistic methodology employed for external flood hazards is a departure from the analysis conducted for other external events such as seismic, hurricane and tornado. The method is somewhat ad hoc in the sense that a complete probabilistic hazard assessment was not conducted (i.e., uncertainty in key parameters are not considered, and a family of flood elevation hazard curves was not produced.) Although the state-of-the-art in flood hazard assessment is sufficiently developed to conduct such an analysis, external flooding in the IPPSS is not treated as thoroughly in a probabilistic context as other external events.

We do not agree with the methodology applied in the IPPSS to evaluate external flood hazards at the site. The approach provides point frequency estimates for single events and event combinations rather than considering a full complement of event sizes, parameter values, and joint occurrence of events. Therefore, at a given frequency of exceedance the uncertainty in flood depth cannot be evaluated, nor can the probability distribution on frequency. We recognize that a reason for this approach is due in part to the traditional notion of a probable maximum flood (PMF). Since the PMF is an extreme event, an annual frequency of occurrence is typically not determined by hydrologists, nor is the variability in key parameters considered. Nonetheless, the uncertainties in estimated frequencies of extreme events are generally considered to be large (Ref. 9). Similarly, for a given storm, there are important sources of uncertainty to be considered in the estimation of flood water surface profiles. The IPPSS has not conducted a sensitivity analysis nor has an analysis been conducted to obtain the uncertainty in the frequency of exceedance. In our judgment a flood hazard analysis should be conducted that accounts for modeling variability and the variability in key parameters of the flooding process.

2.7.3.2 Internal Flooding

In this section the results of an analysis to consider the effects of internal flooding on safety related equipment is considered. At a meeting with IPPSS personnel, a summary was provided of the procedure used to identify sources of internal flooding and to determine their effect. Three steps were followed:

1. Identification of the sources of flooding.

2. Identify locations vulnerable to floods from those sources determined in 1.

3. Cause of initiating event and evaluate the impact.

We generally agree that these are the basic steps required to conduct an internal flood analysis. We would suggest that the internal flood analysis be conducted in a manner suggested in Reference 9 which recommends development of flood analysis fault trees. This would ensure that a thorough, systematic analysis of critical events and event sequences that may lead to a transient are considered. We suspect, based on references in the text, that existing fault trees have been used to some degree in the analysis. However, it is not clear that the effects of localized damage were included in the existing fault trees.

1) <u>Noncategory I Systems</u>

An analysis was undertaken to consider the impact of internal floods on the core melt frequency. The IPPSS study conducts the analysis for Indian Point Unit 2, and based on the similarities in the design of Units 2 and 3, it was assumed that conclusions reached also apply to Unit 3. This assumption is reasonable if it can also be assumed that age effects, particularly in locations where corrosion is likely, do not impact on the results. Also, since the units are not under the same ownership, it should also be verified that conditions have remained the same for both units. Since changes always take place, it is not apparent that equivalent alterations occur at the same time and in the same way in both units. Similarly, temporary blockage of flood passages will undoubtedly be different for each unit. These factors should be addressed in order to verify that the two units are the same. Unless significant changes between them are identified we judge that the difference in the contribution to plant risk will be small.

The IPPSS considered the impact of failure of Noncategory I systems on safety systems. The conclusions reached are based on extensive review by the utility and the NRC (Ref. 10, 11, 12, 13, and 14). The conclusion of the analysis is that the operation of safety systems will not be affected by flooding produced by failure of Noncategory I systems.

a) Circulating Water Failure

A review of flood scenarios is presented due to a circulating water pipe failure. The situations described have been reviewed by the NRC staff. We note that flooding due to a pipe failure is considered to be self-limiting because the condensate pump motors and the 6.9 kV switchgear will be flooded, resulting in reactor trip and loss of offsite power, respectively. This logic presumes that failure events can be counted on to limit the event. The basis for this should be further qualified.

Although a relatively high value for pipe failure is assumed, and no advantage is taken of operator corrective actions, consideration should also be given to potential incorrect action by the operator. Given the high value taken for a pipe failure, the effect of these factors is considered to be small.

b) Fire Protection System Failure

Electrical Tunnel Flooding: Conditions for flooding due to failure of the fire protection system are described. The basis of this event is reasonable; however, no information is provided regarding how the frequencies of valve and pipe failure were determined.

Diesel Generator Building Flooding: We agree with the conclusion that the frequency of diesel generator failure is negligible compared to other causes of failure. However, it is not clear that the frequency of inadvertent accuation has been considered. We judge that considering of this event will have a relatively small effect on the frequency of diesel generator failure.

Charcoal Filter Flooding: We agree with the conclusion and have no additional comment.

2) Category I Systems

a) Primary Auxiliary Building (PAS)

The analysis of flooding in the PAB has been conducted in a manner that identifies the effect of flooding due to the RWST, the service water system and component cooling system. For each system, the frequency of failure has been quantified and considered in the system fault trees. These frequencies are not quantified in the section on internal flooding. The approach taken in the IPPSS is to identify the events that would occur in the event a flood were to occur. It is not apparent from the discussion that the impact of flooding was included in the system fault trees.

b) Diesel Generator Building (DGB)

Flooding in the DGB can be contained by the pit areas and the 12" drain-lines which drain to the circulating water discharge tunnel. Since a plant transient does not occur due to the diesel generators failing, the only event of interest is the joint occurrence of this event and a plant transient. The frequency of this event has been treated in the failure of the service water system. We agree with the conclusion that the likelihood of this event is small.

c) Auxiliary Feed Pump Building (AFPB)

The AFPB has been designed to discharge water from a feedwater line break. However, flood discharge rates of a feedwater line break and drainage capacities are not quantified and, therefore, this statement cannot be evaluated. Our review of this section and Reference 14 and verification that the appropriate failure frequencies are quantified for the auxiliary feedwater system, we have no further comment on this section.

d) Control Building (CB)

Flooding in the CB due to a service water break is considered. Of vital importance is the 480 V switchgear located at level 15'. The analysis assumes that floor drains in the CB will remain available in the case of a flood. To fully demonstrate this, the location of floor drains with respect to the service water lines and the 480 V switchgear must be provided. The conclusion is made that the frequency of power loss is less than the frequency of loss due to other causes. It should be demonstrated in the IPPSS report that the additional increase in the loss of power is negligible.

e) Containment Building

The recent experience of flooding in the containment building has led to significant changes in both units. The numerous changes which have been made are listed in the IPPSS report. No quantification has been made of the frequency of flooding and damage in the containment for the upgraded facility. The reason for this is apparently that a service water system rupture and a LOCA must occur, in order to contribute to plant risk. Due to past experience, a quantification of the system reliability is called for, such as a comparison between the upgraded plant and the system at the time of the 1980 accident. We, in general, agree that the changes have increased the system reliability and that the contribution to plant risk is less than the original design.

We generally agree with the steps performed in the analysis. However, the steps are not given in the IPPSS but were provided at the meeting with IPPSS personnel. We recommend that the methodology and procedures applied be described in the IPPSS report. In future PRAs, we would recommend the use of a more systematic approach, such as Reference 9 recommends.

2.7.4 Fire

For Indian Point, Units 2 and 3, the Indian Point Probabilistic Safety Study (IPPSS) reports that fire accident sequences constitute a significant portion of the overall public risk of plant operations. Based on our review of the Indian Point fire analysis, we have found no evidence which contradicts the IPPSS conclusion that the risk of fire is significant. In fact, it appears that for some accident scenarios the potential importance of fire may be even larger than estimated in the Indian Point fire analysis, primarily because of uncertainties associated with defining the mechanisms by which fire damage occurs (e.g., actual burning or high temperatures). This point will be discussed more thoroughly later in this section.

Scope of Review

Our review of the Indian Point fire analysis considered each fire analysis step identified in the IPPSS. To help resolve initial questions prompted by our review of the fire analysis, we met with the authors of the analysis, and we inspected critical fire areas at Indian Point, Units 2 and 3. Based on this initial review effort, we concluded the following:

1) The Indian Point fire analysis appears to have identified all critical plant areas where a fire can cause an initiating event and, simultaneously, fail redundant safety systems.

2) The Indian Point fire analysis has adopted the best available data base for estimating the frequency of fires in nuclear power plant areas.

3) The Indian Point fire analysis appears to have identified all important safety system components and cabling which are located in critical plant fire areas (See 1).

4) The Indian Point fire analysis reflects as-built plant conditions at the time the analysis was performed; however, subsequent plant modifications to comply with Appendix R to 10 CFR 50, "Fire Protection Program for Operating Nuclear Power Plants," are not included in the IPPSS.

5) The Indian Point fire analysis did not quantitatively assess the importance of a control room fire, even though an analytical basis for excluding the control room from analysis appears to be missing.

Consistent with these initial conclusions, we limited the scope of further review to the manner in which fire scenarios involving specific plant safety areas had been analyzed and quantified in the IPPSS. We did not reassess the selection of critical fire areas, the adaptation of the fire frequency data base, the identification of safety system components and cabling, or the impact of plant modifications implemented subsequent to the original IPPSS analysis. Furthermore, we did not attempt to estimate the safety significance of a control room fire.

The Fire Analysis Method

As stated in the Indian Point fire analysis, "The occurrence of fires and their effects on plant safety are very complex issues which have not received as detailed attention as have other parts of the risk assessment in previous studies. Therefore, major assumptions had to be conservative in order to perform the analysis." In general, these assumptions involved fire occurrence frequencies, fire locations, fire propagation patterns, and fire damage. For each critical fire area in Indian Point, Units 2 and 3, the fire occurrence frequency was estimated using historical data, as represented by a gamma distribution. Within each area, specific fire locations were identified which were judged to constitute the greatest fire vulnerability (e.g., where redundant cable trays cross). The fire occurrence frequency distribution then was reduced to reflect the lower chance that a fire in a critical area will occur at a particular location within the area. Next, the analysis postulated a fire initiation source (e.g., cleaning solvent) and calculated a fire propagation pattern, based on a simplified fire plume model. A Delphi distribution for estimating the time required to extinguish fires was combined with the fire propagation model to arrive at an estimate of the probability that a particular fire will be extinguished before damaging the redundant safety systems of concern. For those fire scenarios which required the subsequent random or operator failure of other safety systems before core melt would occur, the required system unavailabilities were estimated using the same techniques applied elsewhere in the IPPSS.

Based on our review of this analysis method, we have identified two areas of concern which may impact the analysis results. First, the analysis assumes that fire damage occurs only through fire propagation within a fire plume, as a result of fire spreading between fuel zones. Second, the analysis sometimes gives significant credit for successful operator actions, even though the confused operating conditions resulting from a major fire could hamper an operator.

Recent cable fire testing by Sandia has shown that a fire in one part of a room can generate a hot layer of gases along a ceiling, resulting in the failure of cabling located across the room. This testing has indicated that the failed cabling never reached its ignition temperature, but instead, it failed at a lower temperature corresponding to the melting point of its insulation. In the Sandia tests, both IEEE 383 qualified and nonqualified cabling shorted to ground or to other conductors, as a result of hot gas layers generated by both heptane fires and cable fires which were separated from the test cables by twenty feet. The time required to cause shorting ranged from 4.1 to 17.4 minutes depending on the fire configuration investigated. These times are comparable to or less than to the fire plume propagation times estimated in the IPPSS (i.e., 14.4 to 44.0 minutes). If these test results identify fire damage modes which are relevant to Indian Point, then a damaging fire may start in a number of locations within a critical plant area, and the fire may need to be extinguished in a shorter time to assure that redundant systems are not affected by the fire. The Indian Point fire analysis did not consider fire propagation by a hot gas layer or fire damage at temperatures below the ignition point of cabling. The IPPSS cites no data which discounts the possibility of Indian Point cables failing at temperatures below their ignition point.

With regard to giving credit for operator actions, both the Unit 2 and Unit 3 fire analyses have stated that, in the event of a cable spreading room fire, an operator should be able to control auxiliary feedwater pumps locally, by relying on "pneumatic steam generator level indication located inside containment at the airlock." We believe that the conditional mean failure frequency of an operator to achieve safe shutdown by this mode may be higher than the 2.5×10^{-2} value chosen in the IPPSS.

Indian Point, Unit 2

The IPPSS identifies ten different plant damage states and release categories for fire-related scenarios at Indian Point, Unit 2. In terms of IPPSS notation, the ten fire scenarios are:

	Mean Core Melt Frequency
SE/2RW	
Electrical Tunnel (PAB End)	5.6×10^{-5}
Switchgear Room	5.6×10^{-5}
Electrical Tunnel (CB End)	3.2×10^{-5}

TE/2RW	
Cable Spreading Room	3.0×10^{-7}
SLF/8A	
Electrical Tunnel Right Stack (PAB End)	2.4×10^{-5}
Electrical Tunnel Right Stack (CB End)	2.4×10^{-5}
SEF/8A	
Electrical Tunnel Right Stack (PAB End)	1.0×10^{-7}
Electrical Tunnel Right Stack (CB End)	1.0×10^{-7}
TEFC/8B	
Cable Spreading Room	1.6×10^{-6}
SEFC/8B	
Diesel Generator Room	9.0×10^{-7}

In the IPPSS core melt and release frequency summary Table 8.3-9, the SE/2RW fire scenarios are added and, as a result, tie with seismic for first with respect to core melt frequency, while the SLF/8A scenarios rank third in terms of core melt frequency. Because of these high rankings, we discuss these two damage states and release categories more thoroughly in Sections 3.2.2 and 3.2.3 of this report. Before doing this, however, it is interesting to estimate the extent to which the core melt frequencies of the ten fire scenarios may change as a result of considering fire damage by a hot gas layer and placing less reliance on operator actions to achieve safe shutdown.

In order to simplify our reanalysis of the Indian Point, Unit 2 fire scenarios, we will consider only those parts of the analysis which address either the assumed fire location within a critical fire area or the probability of successful operator action to achieve safe shutdown. Parts of the fire analysis involving the frequency of fire occurrence in a critical area or the rate of fire propagation and extinguishment will not be reevaluated, even though these factors include conservatisms which may, in part, balance the nonconservatisms inherent in the fire damage and operator action portions of the analysis.

As a first observation, the damage states and release categories involving a fire in the right cable tray stack of the electrical tunnel (SLF/8A and SEF/8A) would be difficult to postulate in the context of a hot gas layer or cable failure without ignition. Because of the close proximity of the left and right stacks, it may be difficult to distinguish whether a hot layer of gases is generated by a fire on the left or the right or the center of the electrical tunnel. If this is true, then the SLF and SEF damage states should be replaced with a more probable SE/2RW event, corresponding to a fire occurring anywhere in the tunnel, not just in the aisle between the cable tray stacks.

A reestimate of SE/2RW can be found by replacing the IPPSS average value for the fraction of electrical tunnel fires that are large and occur in the aisle portion (i.e., $f_A = 2.66 \times 10^{-2}$) with the average conditional frequency of vertical fire propagation (i.e., $Q(\tau_V) = 0.44$). Although $Q(\tau_V)$ strictly applies to the plume fire propagation pattern considered in the IPPSS up one stack of cable trays in the presence of extinguishing capabilities, it may be viewed as conservatively representing the probability of a hot layer formation, because the values estimated for τ_V (extinguishment time) are generally equal to or larger than the hot layer formation times found in Sandia tests. For a fire in the PAB end of the electrical tunnel, f_A can be replaced with $Q(\tau_V)$ to give a reestimate of the mean core melt frequency of about 1×10^{-3}. However, for the electrical tunnel, $Q(\tau_V)$ appears not to reflect the extraordinary individual cable tray fire suppression systems installed in this location. With this suppression system $Q(\tau_V)$ may be reduced by another order of magnitude. By using a value of $Q(\tau_V)$ of 0.044 and setting f_A equal to unity, the reestimated mean core melt frequency for a fire involving the PAB and CB ends of electrical tunnel become about 1.0×10^{-4} and $1. \times 10^{-4}$, respectively.

For the switchgear room fire which also contributes to SE/2RW, a reestimate can be found by replacing the conditional frequency ($f_{SL} = 1.33 \times 10^{-2}$) of a large exposure fire underneath a critical set of cables with a more conservative number reflecting a hot layer cable failure mechanisms. The f_{SL} factor used in the Indian Point study corresponds to a large fire occurring in a particular portion of the switchgear room floor. One way of viewing this f_{SL} factor would be to consider only one in ten fires in the switchgear room as being big enough to cause a problem and that in order for such large fires to cause significant damage, they must occur in a particular 13 percent portion of the floor area. Another way of viewing f_{SL} would be to consider only 3 out of 200 fires in the switchgear room as being big enough to cause a problem, even if they occur anywhere within the switchgear room. Still, another approach would be to consider any reportable fire in the switchgear room as being big enough to cause significant damage, provided the fire occurs in a particular 1.3 percent portion of the floor area. Although the IPPSS does not clearly state which interpretation of f_{SL} corresponds to the analysis assumptions, it appears that the first interpretation (i.e., one of ten fires being significant can best approximate the situation in the most fire vulnerable areas of Indian Point, Unit 2. This implies for the switchgear room that damaging fires must occur in a particular 13% portion of the room's floor area, assuming cables are burned by an exposed fire. However, if cables fail without burning, as a result of a hot gas layer, then fires occuring over a larger floor area may cause significant damage. If, for instance, fires occurring within 50 percent of the switchgear floor area can cause significant damage by hot layer degradation of cabling, then the switchgear room fire contribution to SE/2RW would be around 2×10^{-4}.

For the diesel generator room fire which contributes to the SEFC/8B damage state and release category, the Indian Point analysis assumed that only two diesel generator fires out of a data base of sixteen fires would have been large enough to cause significant damage at Indian Point. We believe that this assumption may be nonconservative, because a disproportionate number of the fourteen historically "small" diesel fires most likely took place while operators were test starting diesel generators. Under these conditions operators would have been present to extinguish a "small" fire, thereby preventing the occurrence of a larger fire. However, when diesels are automatically demanded, following a loss of offsite power, operators could not necessarily be expected to detect and extinguish an incipient fire. Conservatively one could expect any one of the sixteen historical diesel generator fires to become "large".

With regard to a hot gas layer failure mechanism, the conclusion that only a fire involving the middle diesel generator could damage the other two generators may be invalid. It may be possible to show that a large fire starting at any diesel can generate enough of a hot gas layer to fail the cabling which serves the unburning diesels.

Taking into account all sixteen historical diesel fires and a possible hot layer failure mechanism of all three diesel generators, the following reestimate of the mean core melt frequency of SEFC/8B from a diesel fire can be made using IPPSS values:

Loss of offsite power per year (0.18)	x	Loss of offsite power (including the Unit 1 gas turbine) for at least 60 minutes, as reestimated in Section 3.2.4 of this report (5×10^{-2})	x	Diesel generator mean fire prob ability per demand times three diesel generators (2×10^{-3})	=	1.8×10^{-5}

For the cable spreading room fire which contributes to the TE/2RW and TEFC/8B damage states and release categories, a hot layer failure mechanism may increase the portion of the room floor area in which a damaging large fire (i.e., one in ten fires) may occur from 26 percent to a larger percentage (e.g., 50 percent). In addition to this increase, the estimated frequency of the TE/2RW and TEFC/8B events may increase further, if the average failure frequency of an operator to achieve safe shutdown using local AFW controls proves to be greater than the 2.5×10^{-2} assumed in the IPPSS (e.g., 10^{-1}). The effect of these more conservative assumptions regarding a cable spreading room fire would be a factor of eight increase in the values estimated for TE/2RW and TEFC/8B.

The following table summarizes for Indian Point, Unit 2, the estimated potential impact of assuming a hot layer failure mechanism and a higher operator failure probability for achieving safe shutdown. We have not attempted in these estimates to provide a rigorous reassessment of the Indian Point, Unit 2 fire analysis, and therefore, caution should be used when quantitatively interpreting or directly comparing the reestimated values with other core melt estimates in the IPPSS. Our purpose in making this reassessment was to examine the sensitivity of the Indian Point fire analysis results to the fire scenario assumptions invoked in the analysis.

	IPPSS Mean Core Melt Frequency	Reestimated Mean Core Melt Frequency
SE/2RW		
Electrical Tunnel (PAB End)	5.6×10^{-5}	$\sim 1.0 \times 10^{-4}$
Switchgear Room	5.6×10^{-5}	$\sim 2.0 \times 10^{-4}$
Electrical Tunnel (CB End)	3.2×10^{-5}	$\sim 1.2 \times 10^{-4}$
TE/2RW		
Cable Spreading Room	3.0×10^{-7}	$\sim 2.3 \times 10^{-6}$
SLF/8A		
Electrical Tunnel Right Stack (PAB End)	2.4×10^{-5}	Included In SE/2RW
Electrical Tunnel Right Stack (CB End)	2.4×10^{-5}	Included In SE/2RW
SEF/8A		
Electrical Tunnel Right Stack (PAB End)	1.0×10^{-7}	Included In SE/2RW
Electrical Tunnel Right Stack (CB End)	1.0×10^{-7}	Included In SE/2RW
TEFC/8B		
Cable Spreading Room	1.6×10^{-6}	$\sim 1.2 \times 10^{-5}$
SEFC/8B		
Diesel Generator Room	9.0×10^{-7}	$\sim 1.8 \times 10^{-5}$
Total Estimated Core Melt Frequency From Fire	2.0×10^{-4}	$\sim 4.5 \times 10^{-4}$

Indian Point, Unit 3

The IPPSS identifies six different plant damage states and release categories for fire-related scenarios at Indian Point, Unit 3. In terms of IPPSS notation, the six fire scenarios are:

	Mean Core Melt Frequency
SE/2RW	
Switchgear Room	3.6×10^{-5}
Cable Spreading Room (Tunnel Entrance)	2.4×10^{-5}
TE/2RW	
Upper Cable Tunnel	7.8×10^{-7}
Cable Spreading Room (North Wall)	3.0×10^{-7}
Switchgear Room	7.0×10^{-7}
TEFC/8B	
Cable Spreading Room (North Wall)	1.6×10^{-6}

In the IPPSS core melt and release frequency summary Table 8.3-10, the SE/2RW fire scenarios are added and rank second with respect to core melt frequency. Because of this high ranking, we discuss this damage state and release category more thoroughly in Section 3.3.3 of this report. Before doing this, however, it is interesting to estimate the extent to which the core melt frequencies of the six fire scenarios may change as a result of considering fire damage by a hot gas layer and by placing less reliance on operator actions to achieve safe shutdown in the event of a cable spreading room fire.

In order to simplify our analysis of the Indian Point, Unit 3 fire scenarios, we will consider only those parts of the analysis which address either the assumed fire location within a critical fire area or the probability of successful operator action to achieve safe shutdown. Parts of the fire analysis involving the frequency of fire occurrence in a critical area or the rate of fire propagation and extinguishment will not be reevaluated.

For the switchgear room fires which contribute to SE/2RW and TE/2RW, a reestimate can be found by replacing the conditional frequency ($f_{SL} = 1.33 \times 10^{-2}$) of a large exposure fire underneath a critical set of cables with a more conservative number reflecting a hot layer cable failure mechanism. The f_{SL} figure used in the Indian Point study corresponds to a fire occurring in a particular 13 percent portion of the switchgear room floor (assuming one in ten fires is large). See Unit 2 discussion of switchgear fire (SE/2RW). If instead, a sufficient hot layer could be formed by fires in some larger fraction of the floor area (e.g., 50 percent), than the switchgear room fire contributions to SE/2RW and TE/2RW would be around 1.4×10^{-4} and 2.7×10^{-6}, respectively.

For the cable spreading room fires which contribute to the SE/2RW, TE/2RW, and TEFC/8B damage states and release categories, a hot layer failure mechanisms may increase the portion of the room flow area in which a damaging fire may occur from 17 percent to a larger percentage (e.g., 50 percent). In addition to this, the estimated frequency of the TE/2RW and TEFC/8B events may increase further, if the average failure frequency of an operator to achieve safe shutdown using local auxiliary feedwater pump controls proves to be greater than the 2.6×10^{-2} assumed in the IPPSS (e.g., 10^{-1}). The effect of these more conservative assumptions for cable spreading room fires would be a factor of eleven increase in the core melt values estimated for TE/2RW and TEFC/8B and a factor of three increase for SE/2RW.

For the upper cable tunnel contribution to TE/2RW, a reestimate can be found by representing the IPPSS average value for the fraction of electrical tunnel fires that are large and occur in the aisle portion (i.e., $f_A = 2.66 \times 10^{-2}$) with the average conditional frequency of vertical fire propagation (i.e., $Q(\tau_V = 0.44)$. Although $Q(\tau_V)$ strictly applies to the plume fire propagation pattern considered in the IPPSS up one stack of cable trays, it may be viewed as conservatively representing the probability of a hot layer formation. We believe, however, the the value for $Q(\tau_V)$ does not reflect this extraordinary individual cable tray fire suppression system installed in the Unit 3 electrical tunnel. With this suppression system, $Q(\tau_V)$ may be able to be reduced by another order of magnitude to give a reestimated mean core melt frequency for the upper cable tunnel portion of TE/2RW of about 3×10^{-6}.

The following table summarizes for Indian Point, Unit 3, the estimated potential impact of assuming a hot layer failure mechanism and a higher operator failure probability for achieving safe shtudown. We have not attempted in these estimates to provide a rigorous reassessment of the Indian Point, Unit 3, fire analysis, and therefore, the reestimated values should not be quantitatively interpreted or directly compared with other core melt estimates in the IPPSS. Our purpose in making this reassessment was to examine the sensitivity of the Indian Point fire analysis results to the fire scenario assumptions invoked in the analysis.

	IPPSS Mean Core Melt Frequency	Reestimated Mean Core Melt Frequency
SE/2RW		
Switchgear Room	3.6×10^{-5}	$\sim 1.4 \times 10^{-4}$
Cable Spreading Room (Tunnel Entrance)	2.4×10^{-5}	$\sim 7.2 \times 10^{-5}$
TE/2RW		
Upper Cable Tunnel	7.8×10^{-7}	$\sim 3.0 \times 10^{-6}$
Cable Spreading Room (North Wall)	3.0×10^{-7}	$\sim 3.3 \times 10^{-6}$
Switchgear Room	7.0×10^{-7}	$\sim 2.7 \times 10^{-6}$
TEFC/8B		
Cable Spreading Room (North Wall)	1.6×10^{-6}	$\sim 1.8 \times 10^{-5}$
Total Estimated Core Melt Frequency From Fire	6.3×10^{-5}	$\sim 2.4 \times 10^{-4}$

2.7.5 Transportation and Storage of Hazardous Materials

Section 7.7 of the PRA addressed, in essence, two generic hazardous environments: thermal and toxic. Each of these will be discussed in turn. A third generic environment, blast, was mentioned briefly in the context of secondary missiles. This is discussed in Section 2.7.5.3.

2.7.5.1 Thermal Hazards

The hazards to nuclear power plants from large fires involve the potential for significant damage to safety related structures and equipment. Damage to exposed equipment from thermal hazards is generally not of concern as such equipment is not protected against tornado missiles, and hence is not safety related. [1] However, the service water pumps are exposed as discussed in the section on tornado hazards (IPPSS Section 7.5).

Reference 1 considered the potential effects from fires involving truck and rail car quantities of flammable materials. The standoff distances at Indian point are sufficient to reduce the thermal fluxes from such large fires to negligible levels at plant buildings. The heat capacity of large concrete structures is very high. Even the higher heat fluxes from fireballs (as compared to pool fires) would have negligible effects. In summary, the truck and rail transport of flammable materials would pose negligible risk to the plant.

Section 7.3.3 of the PRA assessed the expected probability of a large, rapid spill of flammable materials on the Hudson River. The probabilities quoted are very, very conservative estimates of plant damage, as a large fire at the shoreline would not produce a sufficient thermal flux at the plant buildings to endanger safety related equipment (excepting again, the service water pumps). The quoted standoff distance is about 0.2 miles.

Section 7.7.4 of the PRA assessed the probability of a large leak from a natural gas pipeline located about 400 feet away from the closest safety related structures. A fire from such a large leak would have to burn for several hours before safety related concrete structure might be threatened. Such long exposures to high heat fluxes do not result in catastrophic failure of structures, but rather in the (conservative) thermal design criteria for reinforced concrete structures being exceeded.

Thus, the probability of 5 x 10^{-7}/year developed in IPPSS Section 7.7.5 is a very conservative estimate for the loss of safety-related equipment. Based on this probability, the contribution to the risk arising from the failure of these exposed pumps due to offsite fires would be expected to be less than that due to tornado hazards. An expected probability of exceeding Part 100 exposure guidelines or of a core melt would be much smaller.

In summary, the probability of thermal fluxes from large fires endangering the safety related structures and equipment is bounded by the failure of this equipment by tornado hazards. The already low probabilities of occurrance of the fires would be very conservative estimates of the probabilities for exceeding Part 100 guidelines or for core melt.

2.7.5.2 Toxic Hazards

Of the chemicals listed in IPPSS Tables 7.7-1 and 7.7-3, only chlorine, anlydrous ammonia and hydrogen cyanide need be considered further. Large releases of these chemicals could lead to incapacitation of the control room operators. Accidents involving the remaining chemicals on the lists would not lead to significant airborne concentrations such that the control room operators could be endangered.

The ongoing analysis mentioned in Section 7.7.6 for chlorine, ammonia and hydrogen cyanide would indicate the level of protection needed for the control room operators.

2.7.5.3 Blast Hazards

In Section 7.7.4, a pipeline explosion was cited as leading to pipe fragments being propelled about 350 feet. It should be pointed out that such fragments would pose minimal risks to reinforced concrete structures. These tumbling, irregular fragments would penetrate only a very small distance into reinforced concrete structures as compared to the design basis tornado missiles (which the concrete structures are designed to withstand).

In summary, blast fragments would be a negligible threat to reinforced concrete structures and even less of a threat to safety related equipment located inside such structures.

References

1. Bennett, D. E. Finley, N. C., Hazards to Nuclear Power Plants from Nearby Accidents Involving Hazardous Materials - A Preliminary Assessment, Sandia National Laboratories, Albuquerque, New Mexico, SAND80-2334, NUREG/CR-1748, April 1981.

2.7.6 Turbine Missiles

The scope of the review was limited by the brevity of the presented analysis. Very little substantial description of methodology or assumptions was given. Additional information was provided by Harold Perla of Pickard, Lowe, and Garrick, and several specific questions were answered which permitted some understanding of the methodology and assumptions. After the above conversation, a short review of the FSAR, Section 14A, was conducted and some simple calculations were made to confirm that all values presented in the FSAR were conservative.

Mr. Perla stated that very little work had been done on this section. The plan was to wait on a forthcoming report, mentioned in the IPPSS, which is in preparation by Westinghouse. When this report became available, the utility was to complete the PRA. (The status of the report and the subsequent PRA is unknown at this writing.) The information presented in the IPPSS was basically taken from the FSAR.

A brief review of the FSAR to determine the source of the value of 10^{-3} for the probability that the missile strikes safety related equipment confirmed that the value is defendable. Although there are several plausible sets of assumptions which might have yielded results similar to those quoted, i.e., probability of striking safety related equipment $\leq 10^{-3}$, the exact ones were not stated in the FSAR. Even though there is no definitive way of evaluating the work done, it appears that the results are acceptable. A set of simple calculations confirmed that the missile penetration analyses were conservative and that the strike probabilities of $\leq 10^{-3}$ were reasonable.

2.7.7 Aircraft Accidents

The IPPSS section on aircraft accidents was reviewed and the results compared with those from analyses published in the literature for aircraft impact. It was tacitly assumed in the IPPSS and continued here that the airborne, suicidal terrorist was not included in this analysis.

The conclusions regarding frequency of impact are reasonable but conservative in several ways, particularly as it follows the standard review plan and as it takes no credit for terminal maneuvers to avoid hard structures by the pilot. This sort of maneuver is to be expected in at least some crashes. No real discussion of consequences of a crash is included but since light aircraft have a striking probability of only slightly greater than 10^{-7} per year and heavy aircraft less than 10^{-7}, this discussion was probably eliminated with justification. No mention was made on an on-site fire following a crash but this too has a combined probability significantly less than 10^{-7} per year.

The conclusions of the IPPSS seem well justified.

2.7.8 Seismic and Wind Fault Trees/Logic Models

The approach to the system modeling used in the IPPSS started with the determination of the fragility of all major components of the applicable plant safety-related systems and structures. All but those components or structures, which were in the range of possible seismic ground acceleration or were exposed to possibly damaging wind/tornado forces, were then eliminated. The systems/component considered seems to be reasonably complete.

The core melt model is then represented entirely by fault trees which were apparently developed from the event tree/fault tree models used in the internal events analysis by eliminating all events not affected by seismic and wind/tornado failures. The resulting fault trees seem reasonable on this basis, although the IPPSS procedure was not reconstructed. (The seismic and wind/tornado fault trees appear in IPPSS Section 7.2 and 7.5 respectively). The core melt models were then combined with the containment fan cooler and spray system models so that the status of the containment following the core melt could be evaluated. This allowed placement of the various core melt cut sets into release categories.

The IPPSS external event systems analysis was entirely separate from the internal event systems analysis until the application of the site matrix. An alternative method, which considers external and internal failures together, would be to apply the external cause directly to the basic internal event tree/fault tree model. As it is, the IPPSS fault trees may fail to identify some important cut sets involving combinations of external and internal events. This point was discussed with IPPSS personnel. They agreed that cutsets are missed; however, they felt that core melt accidents resulting from combinations of external and internal events are probabilistically small, or of less risk significance, in comparison with solely externally caused accidents. This hypothesis seems reasonable to us since external events generally cause common mode failure of redundant systems with a much higher probability than failure of these systems by non-external event "random type" causes.

To test this hypothesis, we postulated several external initiating events in combination with internal events identified in the IPPSS to have a relatively high random probability. The most significant combination of events identified was a core melt sequence at Indian Point 2 with an approximate frequency of 6 x 10^{-7}. This sequence is initiated by a 0.2 g seismic event (5 x 10^{-4}/Ryr, see IPPSS Figure 7.2-4) followed by a loss of off-site power (.5, see IPPSS Figure 7.2-5), and failure of diesel generators 21 and 23 due to random causes (2.1 x 10^{-3}, see Section 4.3). Due to the occurrence of the external event, neither

3. Accident Sequence Analysis

3.1 Introduction

In this section, selected accident sequence analyses are reviewed. Because of the very large number of sequences considered in the report, it is necessary to focus on a subset. We considered the sixteen Indian Point 2 and twelve Indian Point 3 sequences circled on the attached copies of Tables 8.3-9 and 8.3-10, respectively. These include the sequences which, by the IPPSS estimates, dominate core melt frequency or serious radioactive material release frequency. The plant damage state nomenclature is: S or A denotes small or large LOCA and T denotes transient, E or L denotes early or late core melt, F and C denote fans and spray working, respectively. The release category nomenclature listed is Z-1Q, 2, 2RW, 8A and 8B. These are listed in decending order of the severity of radioactive material release from containment, i.e., Z-1Q is the most severe, and 8B the least.

In the following subsections, we review the IPPSS analyses of these sequences. In particular, we compare the IPPSS estimates to alternative estimates based primarily on the reported IPPSS data and thus evaluate the contribution of the IPPSS assumed prior distributions to their estimates. In some cases, we make reference of newer data sources which were not available at the time the IPPSS was performed. We point out where we disagree with IPPSS's assumptions and models and where we would use different human error probability estimates.

This review also tested the readability and reproducibility of the IPPSS. In several respects, the report was found wanting. The sources of numbers used in the event tree calculations (Sec. 1.3) were difficult to trace because of:

- Incorrect references; e.g., a referenced section sometimes would not contain the information claimed to be there.
- Incomplete references; e.g., a reference to 1.5.2 would actually be to 1.5.2.3.4.1.2
- Nonmatching numerical results; in many cases, late changes in the system reliability estimates were not carried through to the event tree analyses so the numbers don't match. For example, the loss of offsite power dominant accident sequence Table 1.3.5.11b-4 was found to be completely wrong and a new table was supplied to us.
- Unclear or inadequate descriptions of events and the IPPSS modeling of them. Descriptions many times had to be clarified with help from the IPPSS authors.

TABLE 8.3-9

COMPARISON OF CORE MELT AND RELEASE FREQUENCY CONTRIBUTION OF MAJOR SCENARIOS, INDIAN POINT 2

SECT	Rank With Respect to Core Melt	Sequence	Major Plant State/ Release Category	Mean Annual Frequency* (Contribution to Core Melt)	Containment Split Fraction to Latent Effects Release	Mean Annual Frequency of Latent Effects Release	Relative Rank With Respect to Latent Effects Release Frequency	Containment Split Fraction to Early Deaths Release	Mean Annual Frequency of Early Deaths Release	Relative Rank With Respect to Early Deaths Release Frequency
3.2.1	1	Seismic: Loss of Control or Power	SE/2RW	1.4-4	1.0-0	1.4-4	1	2.0-4	2.8-8	3
3.2.2	2	Fire: Specific Fires in Electrical Tunnel and Switchgear Room Causing RCP Seal LOCA and Failure of Power Cables to the Safety Injection Pumps, Containment Spray Pumps, and Fan Coolers.	SE/2RW	1.4-4	1.0-0	1.4-4	2	2.0-4	2.8-8	4
3.2.3	3	Fire: Specific Fires in Electrical Tunnel Causing RCP Seal LOCA and Failure of Power Cables to All MCCs, Safety Injection Pumps, RHR Pumps, and Containment Spray Pumps.	SEF/8A	5.0-5	2.0-4	1.0-8	8	1.1-4	5.5-9	5
3.2.4	4	Turbine Trip Due to Loss of Offsite Power: Failure of Two Diesel Generators RCP Seal LOCA, and Failure to Recover External** AC Power Until After 1 Hour.	SEFC/8B	3.0-5	1.0-4	3.0-9	9	1.0-4	3.0-9	8
3.2.5	5	Hurricane, etc., Wind: Loss of All AC Power Due to High Winds.	SE/2RW	2.7-5	1.0-0	2.7-5	3	2.0-4	5.4-9	6
3.2.6	6	Tornado and Missiles: Causing Loss of Offsite Power and Service Water Pumps or Control Building.	SE/2RW	1.6-5	1.0-0	1.6-5	4	2.0-4	3.2-9	7
3.2.7	7	Small LOCA: Failure of Recirculation Cooling	SLFC/8B	1.3-5	1.0-4	1.3-9	10	1.0-4	1.3-9	9
3.2.8	8	Large LOCA: Failure of Low Pressure Recirculation Cooling	ALFC/8B	1.1-5	1.0-4	1.1-9	11	1.0-4	1.1-9	10
3.2.9	9	Medium LOCA: Failure of Low Pressure Recirculation Cooling	ALFC/8B	1.1-5	1.0-4	1.1-9	12	1.0-4	1.1-9	11

*Shorthand notation meaning 4.0×10^{-3}.
**Offsite AC power or gas turbine generator.

SECT = section of this report

TABLE 8.3-9 (continued)

COMPARISON OF CORE MELT AND RELEASE FREQUENCY CONTRIBUTIONS OF MAJOR SCENARIOS, INDIAN POINT 2

SECT	Rank With Respect to Core Melt	Sequence	Major Plant State/ Release Category	Mean Annual Frequency* (Contribution to Core Melt)	Containment Split Fraction to Latent Effects Release	Mean Annual Frequency of Latent Effects Release	Relative Rank With Respect to Latent Effects Release Frequency	Containment Split Fraction to Early Deaths Release	Mean Annual Frequency of Early Deaths Release	Relative Rank With Respect to Early Deaths Release Frequency
3.2.10	10	Turbine Trip Due to Loss of Offsite Power: Loss of All AC Power (Due to Diesel Failure and Combined Diesel/ Service water Failure), RCP Seal LOCA, and Failure to Recover External** AC Power Until After 1 Hour.	SEFC/8B	6.5-6	1.0-4	6.5-10	13	1.0-4	6.5-10	12
3.2.11	11	Large LOCA: Failure of Low Pressure Safety Injection.	AEFC/8B	5.4-6	1.0-4	5.4-10	14	1.0-4	5.4-10	13
3.2.12	12	Turbine Trip Due to Loss of Offsite Power; Failure of Two Diesel-Generators, RCP Seal LOCA, and Failure to Recover External** AC Power.	SEC/8B	4.4-6	1.0-4	4.4-10	15	1.0-4	4.4-10	14
3.2.13	13	Small LOCA: Failure of High Pressure Injection.	SEFC/8B	3.5-6	1.0-4	3.5-10	16	1.0-4	3.5-10	15
	14	Medium LOCA: Failure of Low Pressure Injection.	AEFC/8B	1.7-6	1.0-4	1.7-10	17	1.0-4	1.7-10	17
	15	Fire: Specific Fire in Cable Spreading Room Causing Loss of All Control Power.	TEFC/8B	1.6-6	1.0-4	1.6-10	18	1.0-4	1.6-10	18
3.2.14	16	Turbine Trip Due to Loss of Offsite Power: Loss of All AC Power (Due to Diesel Failure and Combined Diesel/ Service Water Failure), RCP Seal LOCA, and Failure to Recover External** AC Power.	SE/2RW	1.0-6	1.0-0	1.0-6	5	2.0-4	2.0-10	16
	17	Turbine Trip: Failure of AFWS and Failure of Bleed and Feed Cooling.	TEFC/8B	8.5-7	1.0-4	8.5-11	19	1.0-4	8.5-11	19
	18	Reactor Trip: Failure of AFWS and Failure of Bleed and Feed Cooling.	TEFC/8B	7.9-7	1.0-4	7.9-11	20	1.0-4	7.9-11	20

*Shorthand notation meaning 4.0×10^{-3}.
**Offsite AC power or gas turbine generator.

SECT = Section of this report

TABLE 8.3-9 (continued)

COMPARISON OF CORE MELT AND RELEASE FREQUENCY CONTRIBUTIONS OF MAJOR SCENARIOS, INDIAN POINT 2

SECT	Rank With Respect to Core Melt	Sequence	Major Plant State/ Release Category	Mean Annual Frequency* (Contribution to Core Melt)	Containment Split Fraction to Latent Effects Release	Mean Annual Frequency of Latent Effects Release	Relative Rank With Respect to Latent Effects Release Frequency	Containment Split Fraction to Early Deaths Release	Mean Annual Frequency of Early Deaths Release	Relative Rank With Respect to Early Deaths Release Frequency
	19	Medium LOCA: Failure of High Pressure Injection.	AEFC/8B	7.9-7	1.0-4	7.9-11	21	1.0-4	7.9-11	21
	20	Loss of Main Feedwater: Failure of AFWS and Failure of Bleed and Feed Cooling.	TEFC/8B	7.8-7	1.0-4	7.8-11	22	1.0-4	7.8-11	22
3.2.16	21	Seismic: Direct Containment (Backfill) Failure.	Z/Q	6.8-7	1.0-0	6.8-7	5	1.0-0	6.8-7	1
	22	Turbine Trip: ATWS and Failure of AFWS.	SEFC/8B	6.3-7	1.0-4	6.3-11	23	1.0-4	6.3-11	23
	23	Loss of Main Feedwater: ATWS and Failure of AFWS.	SEFC/8B	5.8-7	1.0-4	5.8-11	24	1.0-4	5.8-11	24
3.2.15	24	Interfacing System LOCA	V/2	4.7-7	1.0-0	4.7-7	7	1.0-0	4.7-7	2

*Shorthand notation meaning 4.0×10^{-3}.
**Offsite AC power or gas turbine generator.

ECT = section of the report

TABLE 8.3-10

COMPARISON OF CORE MELT AND RELEASE FREQUENCY CONTRIBUTIONS OF MAJOR SCENARIOS, INDIAN POINT 3

SECT	Rank With Respect to Core Melt	Sequence	Major Plant State/ Release Category	Mean Annual Frequency* (Contribution to Core Melt)	Containment Split Fraction to Latent Effects Release	Mean Annual Frequency of Latent Effects Release	Relative Rank With Respect to Latent Effects Release Frequency	Containment Split Fraction to Early Deaths Release	Mean Annual Frequency of Early Deaths Release	Relative Rank With Respect to Early Deaths Release Frequency
3.3.1	1	Small LOCA: Failure of High Pressure Recirculation Cooling.	SLFC/8B	8.2-5	1.0-4	8.2-9	8	1.0-4	8.2-9	4
3.3.2	2	Fire: Specific Fires in Switchgear Room and Cable Spreading Room Causing RCP Seal LOCA and Failure of Power Cables to the Safety Injection Pumps, the Containment Spray Pumps, and Fan Coolers.	SE/2RW	6.1-5	1.0-0	6.1-5	1	2.0-4	1.2-8	3
3.3.3	3	Large LOCA: Failure of Low Pressure Recirculation Cooling.	ALFC/8B	1.1-5	1.0-4	1.1-9	9	1.0-4	1.1-9	5
3.3.4	4	Medium LOCA: Failure of Low Pressure Recirculation Cooling.	ALFC/8B	1.1-5	1.0-4	1.1-9	10	1.0-4	1.1-9	6
3.3.5	5	Large LOCA: Failure of Safety Injection.	AEFC/8B	6.4-6	1.0-4	6.4-10	11	1.0-4	6.4-10	7
3.3.6	6	Small LOCA: Failure of Safety Injection.	SEFC/8B	2.8-6	1.0-4	2.8-10	12	1.0-4	2.8-10	9
3.3.7	7	Turbine Trip Due to Loss of Offsite Power: Loss of All AC (Due to Diesel Failure and Combined Diesel/ Service Water Failure), RCP Seal LOCA, and Failure to Recover External** AC Power Until After 1 Hour.	SEFC/8B	2.7-6	1.0-4	2.7-10	13	1.0-4	2.7-10	10
3.3.8	8	Seismic: Loss of Control or AC Power.	SE/2RW	2.4-6	1.0-0	2.4-6	2	2.0-4	4.8-10	8
	9	Medium LOCA: Failure of Low Pressure Safety Injection.	AEFC/8B	1.7-6	1.0-4	1.7-10	14	1.0-4	1.7-10	12
	10	Fire: Specific Fire in the Cable Spreading Room Causing Loss of All Control Power.	SEC/8B	1.6-6	1.0-4	1.6-10	15	1.0-4	1.6-10	13

*Shorthand notation meaning 4.0×10^{-3}.

**Offsite AC power or gas turbine generator.

SECT = Section of this report

TABLE 8.3-10 (continued)

COMPARISON OF CORE MELT AND RELEASE FREQUENCY CONTRIBUTIONS OF MAJOR SCENARIOS, INDIAN POINT 3

SECT / Rank With Respect to Core Melt	Sequence	Major Plant State/ Release Category	Mean Annual Frequency* (Contribution to Core Melt)	Containment Split Fraction to Latent Effects Release	Mean Annual Frequency of Latent Effects Release	Relative Rank With Respect to Latent Effects Release Frequency	Containment Split Fraction to Early Deaths Release	Mean Annual Frequency of Early Deaths Release	Relative Rank With Respect to Early Deaths Release Frequency
3.3.9 11	Tornado and Missiles: Loss of Offsite Power and SW Pumps.	SE/2RW	9.2-7	1.0-0	9.2-7	3	2.0-4	1.8-10	11
12	Loss of Main Feedwater: ATWS and Failure of AFWS.	SEFC/8B	7.7-7	1.0-4	7.7-11	18	1.0-4	7.7-11	16
13	Seismic: Loss of Water Storage Tanks	TEF/8A	7.1-7	2.0-4	1.4-10	16	1.1-4	7.8-11	15
14	Loss of Main Feedwater: Failure of AFWS and Long Term Cooling.	SLFC/8B	5.3-7	1.0-4	5.3-11	19	1.0-4	5.3-11	17
3.3.10 15	Interfacing System LOCA:	V/2	4.8-7	1.0-0	4.8-7	5	1.0-0	4.8-7	1
16	TT/LOP: Loss of All AC, RCP LOCA, Failure to Recover.	SE/2RW	4.8-7	1.0	4.8-7	4	2.0-4	9.6-11	14
3.3.11 17	Tornado and Missile: Loss of Offsite Power and RWST.	TEF/8A	4.1-7	2.0-4	8.2-11	17	1.1-4	4.5-11	18
18	Medium LOCA: Failure of High Pressure Injection.	AEFC/8B	3.8-7	1.0-4	3.8-11	20	1.0-4	3.8-11	19
19	Turbine Trip: Failure of AFWS and Long Term Cooling.	SLFC/8B	3.8-7	1.0-4	3.8-11	21	1.0-4	3.8-11	20
20	Loss of Main Feedwater: ATWS and Failure of Pressure Relief.	SEFC/8B	3.3-7	1.0-4	3.3-11	22	1.0-4	3.3-11	21
21	Turbine Trip Due to Loss of Offsite Power: ATWS, Failure of AFWS.	SEFC/8B	2.4-7	1.0-4	2.4-11	23	1.0-4	2.4-11	22
						⋮			⋮
3.3.12 37	Seismic: Containment Failure	Z-1Q	3.7-8	1.0	3.7-8	7	1.0	3.7-8	2

*Shorthand notation meaning 4.0×10^{-3}.
**Offsite AC power or gas turbine generator.

SECT = Section of the report

Specific instances will be cited in the following sections. To aid the reader, pertinent page copies from the IPPSS are included where appropriate.

3.2 Indian Point 2 Dominant Accident Sequence Review

3.2.1 Seismic: Loss of Control or Power, SE/2RW

The Boolean expression for seismic release category 2RW was checked starting with the fault trees and found to be correct. An integration using .he 11 hazard curves from IPPSS report Sections 7.9.1 and 7.9.2 with the 5 fragility curves from IPPSS report Table 7.2-4 was performed using the same relative weighting as the IPPSS, and a mean frequency value of 0.8 x 10-4 per year was obtained. This compares to the value of 1.4 x 10-4 per year reported by the IPPSS. We believe that the difference is due to differences in the integration procedures used and possibly the lumping of hazard curves into the final family used in the DPD operation. A finer discretation of the hazard and fragility points would probably reduce this difference.

The 2RW seismic sequence is the largest contributor to latent effects. It is dominated by the impact between the Unit 1 and 2 control rooms which has a median damage effective ground acceleration of only 0.27g. It is assumed that, if an earthquake large enough to fail the control room occurs, off-site power and the gas turbine will not be available. The next most significant contributor, the superheater stack falling on the control building, has a median capacity of 0.72g, which is almost larger than the upper-bound cutoff value of 0.8g used on the seismic hazard curves. Thus this component does not contribute much to the frequency of 2RW.

Based on a review of the development of the structural capacities, we believe that the mean annual frequency for 2RW equal to 1.4 x 10-4 per year is a factor of 2 low since we feel that the hazard curves given by D&M are more representative of the Indian Point site.

3.2.2 Fire Involving Electrical Tunnel or Switchgear Room, SE/2RW

This accident sequence which contributes to plant damage state and release category SE/2RW combines separate estimates for the following three different fire scenarios:

	IPPSS Mean Core Melt Frequency
Electrical Tunnel (PAB End)	5.6×10^{-5}
Switchgear Room	5.6×10^{-5}
Electrical Tunnel (CB End)	3.2×10^{-5}
	1.4×10^{-4}

A discussion of each of these fire scenarios follows.

A fire that severely damages any of these critical fire areas can affect the power feed to the charging pumps, the containment spray pumps, the component cooling pumps, the safety injection pumps, the PORVs, MCCS 26A and 26B, and all five containment fan coolers. Also, the normal control cabling for the auxiliary feedwater pumps could be lost. Given the loss of component cooling pumps and all charging pumps, a postulated small LOCA through failure of the reactor coolant pump seals was assumed in the IPPSS analysis. With a small LOCA and a loss of the containment spray pumps and fan coolers, an SE/2RW damage state and release category was assessed to occur.

For each of these fire scenarios, the Indian Point fire assessment applied the fire analysis method described in Section 2.7.4 of this report. As part of our reanalysis of these fire scenarios, we examined the sensitivity of the IPPSS core melt frequency estimates to the type of fire damage phenomena postulated for the fire scenarios. In particular, we considered that cabling may be damaged by a hot gas layer, instead of only by a fire plume, as assumed in the Indian Point analysis.

Based on the limited reanalysis which we performed given the time and information available, we show in Section 2.7.4 that, when a hot layer failure mechanism is considered, the mean core melt frequency for this sequence may be as much as a factor of three higher than the value estimated in the IPPSS, i.e., 4.2×10^{-4}.

3.2.3 Fires Involving Electrical Tunnel, SLF/8A

This accident sequence which contributes to plant changes state and release category SLF/8A combines separate estimates for the following two different fire scenarios:

	IPPSS Mean Core Melt Frequency
Electrical Tunnel Right Stack (PAB End)	2.4×10^{-5}
Electrical Tunnel Right Stack (CB End)	2.4×10^{-5}
	4.8×10^{-5}

Note: Table 8.3-9 of the IPPSS incorrectly lists this sequence as SEF/8A.

A fire that severely damages either of these critical fire areas can affect the power cables for the component cooling pumps, charging pumps, containment spray pumps, and both safety related MCCs, 26A and 26B. Given the loss of component cooling pumps and all charging pumps, a postulated small LOCA through failure of the reactor coolant pump seals was assumed in the IPPSS analysis.

The Indian Point analysis states that since the auxiliary feedwater and the high pressure injection systems would not be affected by a fire in the right stack of cable trays, both of these systems could prevent core melt until approximately ten hours after the fire, when the low head recirculation system is needed. Loss of MCCs 26A and 26B would preclude repositioning the necessary valves inside containment to permit low head recirculation with the result being an SLF/8A damage state and release category accident.

For both of these fire scenarios, the Indian Point fire assessment applied the fire analysis method described in Section 2.7.4 of this report. As part of our reanalysis of these fire scenarios, we examined the sensitivity of the IPPSS core melt frequency estimates to the type of fire damage phenomena postulated for the fire scenarios. In particular, we considered that cabling may be damaged by a hot gas layer, instead of only by a fire plume, as assumed in the Indian Point analysis. Based on this cable failure mechanism, we indicate in Section 2.7.4 that, because of the close proximity of the left and right cable stacks in the electrical tunnel, it may be difficult to distinguish whether a hot layer of gases is generated by a fire on the left or the right or the center of the tunnel. If this is true, then the SLF/8A damage state could not reasonably occur, but instead it should be considered included in the SE/2RW damage state in the previous sequence (Section 3.2.2).

3.2.4 Turbine Trip Due to Loss of Offsite Power: Failure of Two Diesel Generators, RCP Seal LOCA, and Failure to Recover AC Power Until After One Hour, SEFC/8B

This sequence leads to plant state SEFC and release category 8B. It has an estimated rate of occurrence of 3.0(-5)/yr which makes it the dominant internal contributor to Indian Point-2's estimated core melt frequency.

The initiating event (11b) has occurred once in six years at IP-2. By merging this information with their assumed prior distribution, the authors arrive at an estimated occurrence rate (their posterior mean) of .18/yr. (Note though that Table 1.5.1-34 in the data appendix gives .20.) In either case, the IP point estimate is consistent with their data and with industry-wide experience.

Following loss of offsite power (LOP), power is to be provided by three diesel generators. Buses 2A and 3A are supplied by one diesel, bus 5A by a second, and bus 6A by the third. Failure of power at buses 2A, 3A, and 5A, or 2A, 3A, and 6A is assumed to lead to a pump seal LOCA in 30 minutes and a core melt in 60 minutes. Thus, the terms in this sequence probability are the failure of either of two pairs of diesels and the failure to provide power from other sources, primarily onsite or nearby gas turbines or from recovery of offsite power.

Actually, though, component cooling water is not lost as long as bus 5A has power, so we find the assumption that a pump seal LOCA follows the failure of the two diesels powering buses 2A, 3A, and 6A to be unnecessarily conservative. We would thus reduce the estimated sequence rate by a factor of two, everything else being equal.

Section 1.3.2.2 of the IPPSS gives the analysis leading to Indian Point's estimated recovery probabilities. In particular, the estimated probability that power is lost to buses 2A, 3A and 5A for 60 minutes sometime during the six hours following LOP is given as 8.9(-5). Losing power to buses 2A, 3A, and 6A has the same probability, so the estimated sequence rate is given by .18 x 8.9(-5) x 2 = 3.2(-5), apparently within rounding error of the 3.0(-5) given in Table 8.3-9, the summary table. Reconstructing the estimated failure to recover probability is hampered by the fact that the report gives distribution plots or medians, where means are needed for the calculations. Nevertheless, some portions of the analysis can be examined.

Power is unavailable initially if two diesels fail to start or if one is out for maintenance and the other fails to start. Thus,

$$Q_o = H^2 + 2HQ_m.$$

where H denotes failure to start and Q_m denotes maintenance unavailability. The following table compares the IP posterior estimates to alternative estimates based on the IP-2 data alone:

	Posterior	Data
Mean (H)	1.3(-2)	9.4(-3) (4/424)
Var (H)	4.2(-5)	2.2(-5)
Mean (Q_m)	3.0(-2)	2.7(-2)
Var (Q_m)	5.4(-3)	3.1(-5)
Q_o	1.0(-3)	6.0(-4)

Thus the IP estimate of Q_o is slightly conservative relative to the data. Note, however, that only independent failures of the two diesels are considered. If the IP (and Zion) conventional β-factor of .014 times H was added to Q_o, the resulting estimate would be 1.2(-3), not markedly different. Alternatively, Oak Ridge (as part of TAP A-44) has estimated the probability of simultaneous failure of two diesels by 7.0(-4) (see Section 2.4.1.1 of this report for the diesel generator β-factor discussion). Adding this to the maintenance term yields an estimated Q_o of 1.5(-3), again not greatly different.

(Readers of the IPPSS will not find Q_o in the study. In supplementary documentation provided us, a quantity denoted by 1-EPO, representing failure of power to two diesels at time zero, was given as 1.0(-3). The above calculations indicate that we have identified the dominant contributors. Page 1.3-14 of the IPPSS gives a value of 1.4(-3) for the probability of power at bus 6A following LOP, and this might be thought to be equivalent to 1-EPO. It is not, though, because it actually is the probability of being in that electric power state at some time during the six hours after LOP. Additionally, the table of dominant sequences for event tree 11b (Table 1.3.5.11b-4) is incorrect, as is its footnote for the source of the modeling of electric power recovery.)

The recovery of AC power must take into account the probabilities of recovering offsite AC power, the onsite diesels, or starting any one of three gas turbines. Examples of the approximate recovery values used in the IPPSS (applies to both units 2 and 3) follow:

Time After Loss of Offsite Power	Offsite Power Recovery Probability of	Recovery of DGs		Gas Turbine
		2 DG Failure	Blackout	
30 min.	.37	.25	.08	.11
60 min.	.55	.35	.18	.89
90 min.	.68	.43	.25	.99 (applies to longer times)

Comparison with previous assessments of recovery shows that offsite power recovery values are similar to past estimates. DG recovery values appear quite optimistic when compared with past experience. The IPPSS values are based on a critical review of DG failure modes and corresponding times to repair and not on actual experience. In part, this is a valid thing to do since actual experience is based on noncritical AC loss conditions, and therefore times to DG repair were unnecessarily long. However, the optimistic DG recovery values are relatively unimportant relative to the AC recovery potential based on starting a gas turbine.

Therefore, the recovery model is most dependent on the gas turbines. The scenario assumed is that first an attempt to fix and start the diesels will be made. Failing that in 15 minutes, the decision will be made to start the onsite gas turbine. Failing that, an operator will drive one-half mile to the Buchanan substation and attempt to start the two gas turbines there. Families of probability distributions for the time required to perform these tasks were assigned, based on reviews of the steps involved, not actual experience. The resulting degree-of-belief median for the probability of failing to obtain power from a gas turbine is .11. This value can be read from the median curve in IPPSS Figure 1.3.2.2-5. The 5th and 95th percentiles shown correspond approximately to statistical confidence limits based on 20 failures in 170 trials. We doubt that the speculation that went into the IP estimates is "worth" this much data, but we have no basis for an alternative point estimate.

The times to perform each step for reaching and subsequently starting a gas turbine appear reasonable, if not conservative (e.g., 4-15 minutes to drive one-half mile assumes speeds of 2-7 mph). In addition, the failure probability of each gas turbine at $\sim$ 1E-1 also appears consistent with other failure estimates, making the gas turbine recovery factors seem quite reasonable.

Failure to recover offsite power within 60 minutes is estimated to have a probability of .45 (median value). This is consistent with previous estimates (and conservative relative to the value of .26 we assumed in our Zion review). Failure to obtain power from either the gas turbines or offsite within 60 minutes is thus estimated by .11(.45) = .05. This product of medians is somewhat less than the mean of the product .09 which, by multiplying this value of (1-EPO), yields 5.0(-5). This value is just over one-half of the 60-minute power failure probability of 8.9(-5). Plausibly, using mean values and accounting for other than initial failures could make up the difference. We thus find no reason for a markedly different estimate from what Indian Point obtained, except to repeat that the considered bus failure combination is conservative, and the estimated sequence frequency can be reduced by about a factor of two with the inclusion of the bus 5A consideration.

3.2.5 Hurricane, etc., Wind: Loss of All AC Power Due to High Winds, SE/2RW

For hurricane winds, release category 2RW is dominated by the Boolean expression ②W^(④W V ⑥W) where the symbols correspond to offsite power, the control building (which houses the switchgear and batteries for starting the diesel generator), and the diesel generator building, respectively. Other parts of the equation are controlled by tornado missile capacities which are not possible for hurricanes. Unlike analysis presented in the IPPSS, we believe that offsite power should be considered to have failed if a hurricane occurs. Loss of offsite and on-site AC power results in a small break loss of coolant (pump seal LOCA) sequence with no injection and no containment safe-guards. Because of the steepness of the hurricane hazard curves, assuming that offsite power is unavailable, will increase the mean frequency of 2RW by a factor of at least 2. We also believe that the fragility curves may be on the uncon-servative side; however, due to the protection provided by adjacent structures, the implicity assumed shape factor value of 1.0 may have resulted in overpredicting the control room fragility capacity for wind pressure effects.

Based on review of IPPSS Section 7.9.5, we believe that the median hurricane hazard curve is unconservative. A compar-ison of the IPPSS median and upper-bound curves was made with hazard values obtained from Batts, M. E., et al, "Hurricane Wind Speeds in the United States," NBS Building Science Series 124, National Bureau of Standards, May 1980. Using a range of hazard curves based on this reference and the median fragility curve from IPPSS Table 7.5-4, we obtain a factor of 10 to 30 increase in release category 2RW. We believe that a factor of 10 increase is appropriate for differences due to the hazard curves.

In developing the Boolean equation for 2RW, part of the probability of the stack failing and falling on the control or diesel generator buildings was omitted. The capacity of the stack is relatively high and the omission of the stack failing does not significantly effect the frequency of 2RW.

In summary, we believe that the 2RW mean failure frequency value of 2.7 x 10-5 per year for hurricane effects may be low by a factor of 20 due to revised fragility for offsite power and an increase in the hurricane hazard at the site.

3.2.6. Tornado and Missiles: Causing Loss of Offsite Power and Service Water Pumps or Control Building, SE/2RW.

For tornado winds, release category 2RW is dominated by the same Boolean expression as discussed above for hurricanes. Other parts of the sequence equation (i.e., including service water pumps and the RWST) are controlled by tornado missile capacities which are high relative to wind pressure capacities. Assuming that offsite power is not available will not chan, the tornado 2PW frequencies quite as much as for hurricane effects. Because the hazard curves for tornado are less steep than the hurricane curves, it is estimated that, if offsite power is unavailable, the mean value will change by a factor of less than 2. We believe that the tornado hazard curves are conservative and, if decreased based on an estimate which considers the tornado history in the vicinity of the site, would lower the mean value by a factor of 2 to 10. In summary, the mean value of 1.6 x 10-5 per year is reasonable and probably conservative.

3.2.7 Small LOCA: Failure of Recirculation Cooling, SLFC/8B

This sequence leads to the SLFC plant state and release category 8B. A small LOCA is estimated to occur at an annual frequency of .0185/yr, and the recirculation failure probability is estimated as 6.8(-4). The product, 1.3(-5), is Indian Point's estimate of the frequency of this sequence.

The small LOCA frequency is estimated by first estimating the distribution of the small LOCA rate among PWRs. Though the "state-of-knowledge" and data that go into the specification of this "prior" distribution are unchanged from that which went into the Zion analysis, a different prior was chosen here. In this case (in contrast to the large LOCA discussed in Section 2.6), the IP prior was more optimistic; the prior variance was less by a factor of 30. This, plus the fact that IP has had no small LOCAs while Zion has had one, led to the above estimate for IP-2's small LOCA frequency, which is about one-half of the Zion estimate (3.5(-2)). The PWR data alone show no appreciable evidence of plant-to-plant variation and so, if combined, would yield an estimate of 3/131 = .023.

The (high-head, for small LOCA) recirculation failure probability estimate is divided about 60-40 between operator errors and hardware failures, specifically 3.9(-4) and 2.9(-4), respectively. The operator errors are

1) Failure to initiate switchover to high pressure recirculation.

2) Inadvertent actuation of "Switch 7" instead of "Switch 6."

In the IPPSS, the probability of the first event is determined by first estimating the failure probability of one reactor operator (RO) to initiate switchover and then subsequently estimating the conditional probability that three other people (another RO, the watch supervisor who is a senior RO, SRO, and the shift technical advisor, STA) fail to detect and correct the first RO's failure. Let P denote the first RO's failure. The Indian Point's expression for the probability of failure to initiate switchover is

$$Q_{HI} = P \left(\frac{1+P}{2}\right) \left(\frac{1+6P}{7}\right) \left(\frac{1+6P}{7}\right) ,$$

the terms in the product corresponding to RO_1, RO_2, SRO, and STA. For small P (as assumed by IP),

$$Q_{HI} \approx P \left(\frac{1}{2}\right) \left(\frac{1}{7}\right)^2$$

$$\approx .01\ P$$

The initial error probability is estimated (in lognormal terms) as a median value of .006 with an error factor of 5. The conditional probabilities are based on the authors' impression of the levels of dependence among the personnel, translated into numbers using "Swain's Handbook." To reflect uncertainty in these estimates, IP's assumed error factor on Q_{HI} is increased to 20 and the resulting mean probability of Q_{HI} is 3.5(-4). As discussed in Section 2.5 of this report, the dependence level of the other operators is undoubtedly greater than that expressed in the IPPSS, yet at the same time, the study did not account for the recovery potential activated by the RWST low-low level annunciator. Thus, it is believed that the first of the human errors arising in recirculation failure is negligible. (However, it should be noted that the Brookhaven reviewers felt that no credit should be given for the fourth person in the control room and that the initial error probability should be larger, and this led them to an estimated Q_{HI} of 2.8(-3).)

As to the second human error involved in changing to recirculation cooling, the analysis presented in Section 2.5 of this report shows that this, too, is of negligible probability.

Indian Point's hardware failure probability estimate is dominated by their estimate of nonindependent failures of any of four pairs of MOVs or three safety injection pumps. These estimates are based on their standard β-factor of 0.014. Given this assumption, their estimates are reasonably consistent with the available data. As discussed in Section 2.4.1.7, however, a better estimate of the β-factor for the dependence of the three pumps is 0.052. If this is used, the hardware contribution to high-head recirculation failure increases to 8.2(-4) from 2.9(-4).

Thus, in conclusion, it is felt by the reviewers that a better estimate of the frequency of this sequence is 1.5(-5)/yr, instead of the 1.3(-5)/yr value reported in the IPPSS.

3.2.8 Large LOCA: Failure of Recirculation Cooling, ALFC/8B

This sequence leads to plant state ALFC and release category 8B. A large LOCA is estimated (see Section 2.6 of this report) to occur at a rate of 1.95(-3)/yr and low pressure recirculation failure probability is estimated as 5.4(-3), thus leading to an estimated sequence rate of 1.1(-5)/yr.

The determined dominant source of recirculation failure is failure to initiate switchover (over 97 percent of the failure probability). Relative to a small LOCA (see 3.2.7, above), the operators have less time to initiate switchover and are under higher stress. In Section 2.5.4 of this report, the IPPSS modeling of failing to correctly initiate switchover after a large LOCA was reviewed in detail. It was concluded there that the IPPSS underestimated the "8 switch sequence error" by a factor of 30. We therefore feel a better estimate of failing to correctly initiate switchover is ~.02.

Thus, a better estimate of the frequency of this sequence is 3.9(-5)/yr instead of the 1.1(-5)/yr estimate given in the IPPSS.

3.2.9 Medium LOCA: Failure of Recirculation Cooling, ALFC/8B

The analysis of this sequence is identical to that for a large LOCA (Section 3.2.8).

3.2.10 Turbine Trip Due to LOP: Loss of All AC Power, RCP Seal LOCA, Failure to Recover External AC Power Until After One Hour, SEFC/8B

This sequence differs from that discussed in Section 3.2.4 in that power is lost to all buses. This can happen if all three diesels fail or if one or two diesels start, but service water fails, thus failing all the diesels. In supplementary material provided Sandia, the estimate in IPPSS of initial loss of all AC power is 4.0(-4) and the 60-minute failure-to-recover probability is estimated as .08, thus yielding .18 x 3.2(-5) = 5.8(-6)/yr as the authors' estimated sequence rate. (Note: Table 8.3-9 gives 6.5(-6).)

Consider the case of triple diesel failure to start. The approximate model is

$$Q_o = H^3 + 3H^2Q_m .$$

The mean value of Q_o (based on the information in Tables 1.5.2.2.1-10L and -10O) is 2.1(-5), so triple diesel failure is a minor contributor to this sequence relative to the various combinations of diesel and service water failure. Nonindependent failures are considered to be negligible. Even if the TAP A-44 estimate for the simultaneous failure of three diesels, namely 3.0(-4), is added in, the sequence estimate is not markedly changed.

The diesel generator/service water interactions were examined, and with the use of the IPPSS assumption and methods, the failure probability of 4.0(-4) given above for the initial loss of all AC power was confirmed. There were two problems identified with the analysis presented in IPPSS, however. First, the service water values used for the interaction were for the entire system, both the nuclear and conventional headers, whereas the diesel generators receive cooling from only the nuclear header. This conservatism is slight, though, because the conventional header failure probability is generally small with respect to the nuclear header.

The second problem was the use of the nuclear header success criterion for this sequence. Although the IPPSS description states that for the cooling of all three diesel generators, only one nuclear header pump is necessary, the actual criterion used was the same as for other sequences, i.e., that two nuclear header pumps were necessary. This discrepancy surfaced in discussion with the analysts. Subsequently, data has been supplied for the one pump criterion which indicate that the 4.0(-4) probability should be 1.7(-4) instead. (It must be noted that although the diesel generator dependency on nuclear header service water was initially incorrectly used, the electric power dependency of the nuclear header service water was correctly analyzed.)

For other sequence considerations, the nonrecovery factor of 0.08 is consistent with that discussed in our Section 3.2.4 (as it should be); so our comments there apply here. Overall, with the service water success criterion relaxed for this sequence, the sequence frequency becomes 2.4(-6)/yr.

3.2.11 Large LOCA: Failure of Low Pressure Safety Injection, AEFC/8B

This sequence leads to plant state AEFC and release category 8B. The estimated probability of the failure of low pressure safety injection, labeled LP-1, is 2.8(-3), which when multiplied by Indian Point's estimate of the large LOCA rate (1.95(-3)/yr), yields an estimated sequence rate of 5.4(-6)/yr.

The source of the LP-1 estimate is given in IPPSS Section 1.3.3, which is a section that gives various supporting analyses. That for LP-1 is missing, though. In Section 1.3.4.1.2, which discusses the large LOCA event tree, LP-1 is defined as failure of either the low pressure injection or accumulator system. The analyses for these two systems in Section 1.5 give estimated failure probabilities of 8.7(-4) for low pressure injection and 1.9(-3) for the accumulators, which, when summed, yield the LP-1 estimate of 2.8(-3). An examination of the bases of this estimate, and supporting data from Zion as well as Indian Point, provides no reason to choose markedly different estimates, although further analysis can yield some change.

The IPPSS does not give a variance or any percentile associated with the authors' state of knowledge distribution for LP-1. The variance, though, can be derived from the information given. The accumulators fail if any of six check valves or three MOVs fail. The former are treated as demand-dependent and the latter are time-dependent with a half-test interval of 9 months = 6570 hours. Thus,

$$Q_{ACC} = 6\ P_{CV} + 3\ (6570)\lambda_{MOV}\ ,$$

where P_{CV} denotes the check valve failure probability, on demand, and λ_{MOV} denotes the hourly failure rate of MOVs. The single failure and dominating term in the failure of low pressure injection are two check valve failures, two MOV failures, and one manual valve failure. Two of these valves are tested monthly, the other at 18 months, and manual and motor-operated valves are assumed to have the same failure rate. Thus,

$$Q_{LP-1} \approx 2\ P_{CV} + 7290\ \lambda_{MOV}\ .$$

Adding yields

$$Q_{LP-1} \approx 8_{PCV} + 27{,}000\ \lambda_{MOV}\ .$$

IP's posterior mean and variance for the two right-hand parameters are:

	Mean	Variance
P_{CV}	7.0(-5)	1.1(-8)
λ_{MOV}	7.4(-8)	5.9(-15)

Thus, the variance of Q_{LP-1} is (at least, since some failure modes have been omitted)

$$\text{var}\left(Q_{LP-1}\right) = 64 \times 1.1(-8) + (27{,}000)^2 \times 5.9(-15)$$

$$= 5.0(-6)$$

The mean value of Q_{LP-1}, considering just single failure, is 2.6(-3); so, using the methodology of Section 2.6, IP's estimate of LP-1 corresponds to effective data of 1.4 failures in 520 demands.

If just the data from Indian Point 2 are considered, there have been no failures of either type, and the component data can be reduced to effective data for LP-1 of 0/180. The addition of IP-3 and Zion data (also no failures) leads to effective LP-1 data of 0/770. Hence, "adding in" IP's prior distribution has a more conservative effect than adding in consistent experience from two other units. Based on these considerations, we find no cause for an appreciably different estimate than that given by the IPPSS for the valve failures.

Two other comments on the analysis must be made. The first is, as discussed in Section 2.4.1.6 of this report, the upstream check valves in the injection piping are counted twice in the IPPSS quantification, once for the accumulators and once for LP injection. Since the failure of either system fails coolant injection, the effect of this modeling is to overestimate the failure probability of coolant injection. IPPSS gives the mean failure of three check valves failing to open as 2.1(-4).

The last comment concerns possible dependencies between the two LP pumps. IPPSS uses their general β-factor of 0.014. As presented in Section 2.4.1.5 of this report, a β-factor of 0.080 is deemed to be more appropriate. If this value is used, the failure probability of low pressure injection becomes 1.2(-3).

Hence, with the use of this β-factor and discounting the double entry of the three check valves, a better estimate of the frequency for this sequence is 5.6(-6)/yr, an overall increase of just 4 percent.

3.2.12 Turbine Trip Due to LOP: Failure of Two Diesel Generators, RCP Seal LOCA, Failure to Recover AC Power (Within Three Hours), SEC/8B

This sequence differs from that discussed in our Section 3.2.4 only in that recovery is not until after three hours rather than one. It leads to plant state SEC, rather than SEFC, but the same release category, 8B.

The assumed AC recovery distributions lead to a three-hour failure of AC power probability of 1.4(-5), which is about 1/7 the estimated one-hour value of 8.9(-5). Such a ratio seems plausible when compared with the recovery potential difference in the gas turbines from one to three hours (see recovery probability table presented in Section 3.2.4). Since the greatest recovery change is in the gas turbines, which improves by a factor of ~ 10, a sequence frequency reduction factor of 10 compares favorably with the factor of 7 just mentioned above.

Also, as discussed in Section 3.2.4, the authors assumed there are two pairs of diesel failures that could lead to this core melt sequence, while we conclude that only one pair does so. Thus, we would halve this sequence estimate. Overall, we have no basis for a markedly different estimate from that given by Indian Point.

3.2.13 Small LOCA: Failure of High Pressure Injection, SEFC/8B

This sequence results in plant state SEFC and release category 8B. The small LOCA rate is estimated as .0185/yr (see Section 3.2.7). High pressure injection fails if any of the three suction valves from the RWST fails or if two (of two) safety injection pump trains fail. The estimated probability of system failure is 1.9(-4), two-thirds of which comes from the RWST single valve failures, the remainder from dual failures in the pump trains.

The valve failure probability estimates are consistent with the available data (no such failure at Zion or Indian Point). Common cause failures are not considered by IP (apparently an oversight because for the same equipment following a medium LOCA, a ß-factor of .014 was assumed). By including this ß-factor, Indian Point's system failure estimate would increase to 2.8(-4). Brookhaven's reviewers argue that accounting for common cause failures and estimating pump failure probability less optimistically (IP's prior is considered more optimistic than the plant specific data, apparently because different failure modes were considered) indicate a system failure probability of 1.5(-3), which is a factor of eight times the IP estimate.

As presented in Section 2.4.1.4., a more appropriate ß-factor is 0.051 which, when combined with the rest of the IPPSS data for high pressure injection, yields a system failure probability of 5(-4). This then would result in a sequence frequency of 9.3(-6)/yr, an increase by a factor of about 2.5.

3.2.14 Turbine Trip Due to LOP: Loss of All AC Power, RCP Seal LOCA, Failure to Recover AC Power (Within Three Hours), SE/2RW

This sequence leads to plant damage state SE and release category 2RW. Its estimated rate of occurrence of 1.0(-6)/yr makes it the leading internal event in terms of the risk of latent cancers.

This sequence differs from that discussed in Section 3.2.10 only in that recovery of AC power is after three hours, rather than one. As discussed in Section 3.2.12, the probability of no recovery for three hours is estimated to be about one-seventh that for one hour, which seems plausible, based on changing the gas turbine recovery estimates from one to three hours. Because of the corrected service water criterion presented in Section 3.2.10, the frequency of this sequence should decrease by about a factor of two, i.e., 5×10^{-7}.

3.2.15 Event V: The Interfacing Systems LOCA, V/2

The internal event which dominates risks, in terms of early fatalities, according to the Indian Point estimates, is the interfacing systems LOCA--a LOCA that bypasses containment. The dominant V sequence is the joint failure of two motor-operated valves in the RHR suction path. A description of this event and the resulting estimates are given on pages 1.3-241,242 for IP-2 and 1.3-448,449 for IP-3. Conversations with the authors indicate that this description and the calculation which accompanies it are "inoperative."

The real situation is apparently this: After a refueling outage both valves are supposed to be closed. One valve may not be because of an undetected failure when the valve is demanded closed, but at least one valve must be closed in order to have a successful startup. In the subsequent (assumed) 18 months between refuelings, Event V can happen if one valve was not closed at startup and the other ruptures or if both valves were closed, but then the upstream, followed by the downstream, valve ruptures. At our request, IPPSS personnel performed a revised analysis. We reviewed their analysis and based on the information presented, found it acceptable.

The revised IPPSS model is

$$P(V) = 2p\lambda T + [1-e^{-\lambda T}(1+\lambda T)]$$

where P = probability of valve failure to close on demand in an undetected manner

λ = valve rupture failure rate (hr^{-1})

T = time between refuelings (hr)

The values of P and λ used in the calculation do not appear in the IPPSS. They were derived, at our request, from an extensive analysis derived from "Nuclear Power Experience" data. The values of P and λ derived were

$P = 5.8 \times 10^{-5}$ (IP2); 3.8×10^{-5} (IP3)

λ =	1.2×10^{-8}	mean	
	2.2×10^{-9}	median	
	3.7×10^{-15}	variance	(IP2 and IP3)
	1.1×10^{-10}	5 percent	
	4.4×10^{-8}	95 percent	

Substitution of these values in the model and performance of the discrete probability distribution arithmetic yields the following:

IP2	P(V) =	3.4×10^{-7}	mean
		3.4×10^{-9}	median
		3.1×10^{-11}	variance
		3.1×10^{-11}	5 percent
		6.1×10^{-7}	95 percent

IP3	P(V) =	4.6×10^{-7}	mean
		2.2×10^{-9}	median
		4×10^{-11}	variance
		2.4×10^{-11}	5 percent
		6.5×10^{-7}	95 percent

The mean values are dominated by the second term in the model which represents the rupture of both valves.

The revised means are not significantly different from he means appearing in the IPPSS. The IPPSS distributions, however, are conservative with respect to the revised distributions.

3.2.16 Seismic: Direct Containment (Backfill) Failure, Z-1Q

The sequence leading to release category Z-1Q consists entirely of shear failure of the containment building wall. Because of the relatively high capacity for this failure mode (i.e., median value equal to 1.1g) the mean frequency of failure is only 6.8×10^{-7} per year. The frequency of Z-1Q is sensitive to the upper-bound cutoff on the hazard curves. Because we feel that the D&M hazard curves are more representative, the frequency of Z-1Q is a factor of 2 low. The reason that the frequency of release for category Z-1Q is higher for Unit 2 compared to 3 is due to the large soil loading on the Unit 2 containment building.

3.3 Indian Point 3 Dominant Accident Sequence Review

3.3.1 Small LOCA: Failure of High Pressure Recirculation, SLFC/8B

This sequence leads to plant state SLFC and release category 8B. By IP's estimates, it is the most likely cause of core melt at IP-3. A small LOCA is estimated to occur at a rate of .020/yr, and the estimated recirculation failure probability is 4.1(-3), thus yielding an estimated sequence rate of 8.2(-5)/yr. By way of contrast, for IP-2 the estimated recirculation failure probability was 6.8(-4), thus yielding a sequence rate of 1.3(-5)/yr. The systems (in the two units) appear similarly designed, and the data are consistent, so this much of a difference seems surprising. In fact, it is an artifact due to an over-analysis of the data.

The source of the difference between the IP-2 and IP-3 estimates is the experience of the safety injection pumps. IP-2 shows zero failures to operate in 84 hours; IP-3 shows 1/40 hours. When this fairly meager experience is merged with the assumed prior distribution (which has a mean of 2.0(-5)/hr), the corresponding posterior means are 1.6(-5)/hr and 1.8(-3)/hr, two orders of magnitude apart. Thus, an independent triple failure to run 24 hours is the dominant estimated failure for IP-3, negligible for IP-2.

If it were argued in the IPPSS that the SI pumps were markedly different in, say, manufacturer or operating procedures between the two units, one might accept different estimates. In fact, one might claim the IP-3 result is an underestimate because of the effect of the very optimistic prior. Without that argument, there is little reason to assume different failure rates for SI pumps at the two units and one would be led to combine the data, thus estimating the failure rate from data of 1/124 hours. When combined with IP's prior, the result would be sequence estimates for both units in the neighborhood of the IP-3 estimate.

On the other hand, the IP-3 failure that was counted is based on a quite conservative interpretation of the LER. The pump did not fail, but was repaired because of degraded performance. Thus, we would discount this failure and accept the IP-2 estimated sequence rate of 1.5(-5) as the starting point for further examination. As presented in Section 3.2.7, the operator errors considered by IPPSS have been evaluated to be negligible. Thus, with the use of the adjusted system hardware failure given in Section 3.2.7, we conclude that the frequency of this sequence is approximately 1.5(-5)/yr.

3.3.2 Fires Involving Switchgear Room or Cable Spreading Room, SE/2RW

This accident sequence which contributes to plant damage state and release category SE/2RW combines separate estimates for the following two different fire scenarios:

	IPPSS Mean Core Melt Frequency
Switchgear Room	3.6×10^{-5}
Cable Spreading Room (Tunnel Entrance)	2.4×10^{-5}
	6×10^{-5}

A fire that severely damages either of these critical fire areas can affect the power cables for the charging pumps, the containment spray pumps, the component cooling pumps, the safety injection pumps, and all five containment fan coolers. Given the loss of component cooling pumps and all charging pumps, a postulated small LOCA through failure of the reactor coolant pump seals was assumed in the IPPSS analysis. With a small LOCA and a loss of the containment spray pumps and fan coolers, an SE/2RW damage state and release category was assessed to occur.

For each of these fire scenarios, the Indian Point fire assessment applied the fire analysis method described in Section 2.7.4 of this report. As part of our reanalysis of these fire scenarios, we examined the sensitivity of the IPPSS core melt frequency estimates to the type of fire damage phenomena postulated for the fire scenarios. In particular, we considered that cabling may be damaged by a hot gas layer, instead of by a fire plume, as assumed in the Indian Point analysis.

Based on the limited reanalysis which we performed given the time and information available, we show in Section 2.7.4 that, when a hot layer failure mechanism is considered, the mean core melt frequency for this sequence may be as much as a factor of 3.5 higher than the value estimated in the IPPSS, i.e., 2.1×10^{-4}.

3.3.3 Large LOCA: Failure of Low Pressure Recirculation Cooling, ALFC/8B

IP's estimate and analysis for this sequence are negligibly different from that for Unit 2 (see Section 3.2.8).

3.3.4 Medium LOCA: Failure of Low Pressure Recirculation Cooling, ALFC/8B

No difference from IP-2 (Sections 3.2.8 and 3.2.9), and the same frequency presented above in Section 3.3.3 applies.

3.3.5 Large LOCA: Failure of Safety Injection, ALFC/8B

The system model for IP-3 is virtually the same as for IP-2 (see Section 3.2.11). Negligible differences in the plant specific data plus different assumptions about common cause failures yield slightly different sequence estimates: 6.4(-6) for IP-3, 5.4(-6) for IP-2. The analysis presented in Section 2.4.2.5 shows that the LPI failure probability for IP-3 is closer to 9.2(-4) instead of the reported 8.1(-4) value. With this change and removing the included double check valve accounting, the estimated sequence failure probability becomes 6.2(-6).

3.3.6 Small LOCA: Failure of Safety Injection, SEFC/8B

Indian Point's model for IP-3 differs from that for IP-2 (Section 3.2.13) only in that where one MOV in an SI pump train was assumed to be tested every 18 months at IP-2, monthly testing is assumed for IP-3. Also, the plant specific data on pump fail-to-start probability indicates a possible difference between units: At IP-2, there have been seven failures in 793 demands; the IP-3 data are 2/800. The cumulative effect of these two differences (and other more minor ones) is that the sequence estimate for IP-3 is 2.8(-6) versus 3.5(-6) for IP-2. The analysis presented in Section 2.4.2.4 shows that a better failure probability estimate of the safety injection system is 4.9(-4) instead of the 1.3(-4) presented in IPPSS. The difference is attributable to the fact that in the IPPSS three single failures contribute essentially all of the HP injection system failure probability as the system is analyzed. They are check valve 847, motor-operated valve 1810, and manual valve 846 which are all in the common pump suction line from the RWST. The analysis presented in Section 2.4.2.4 of this report suggests that, in fact, the failure of any of these valves is not the dominant contributor to system unavailability. Rather, common cause failure of all three pumps failing to start and run dominates with a failure probability of 3.6(-4).

With the addition of this pump failure mechanism, the sequence frequency is recalculated to be 9.8(-6)/yr.

3.3.7 Turbine Trip Due to LOP: Loss of All AC, RCP Seal LOCA, Failure to Recover AC Power Until After One Hour, SEFC/8B

This sequence leads to damage state SEFC and release category 8B. The model is virtually the same as that for the same sequence at Indian Point-2 (see our Section 3.2.10), but the estimated rate of occurrence is lower: 2.7(-6)/yr versus 6.5(-6)/yr. This difference can be traced primarily to different estimates of service water systems failure probability (recall that various combinations of diesel failures and service water failure lead to loss of all AC). The service water estimates differ primarily because the estimated probability of failure of a pump to start on demand differ.

The IP-2 pump failure data are 7 failures in 753 demands; the IP-3 data are 2/800. When these were merged with the same optimistic prior distribution (optimistic because the prior excluded command faults, the plant specific data included them), the posterior means were 6.4(-3) and 1.4(-3). Under binomial distribution assumptions, the apparent difference between pumps at the two units could easily be due to chance. In this case, the number of demands are estimates and the data have been pooled across various types of pumps, so it is reasonable to combine the data across units. The result is an estimated failure probability of 9/1593 = 5.6(-3), not greatly different from the IP-2 posterior mean. We would thus conclude that the IP-2 estimated sequence rate of 6.5(-6) should also apply to IP-3.

In addition, if the same diesel generator/service water interaction is used as in Section 3.2.10, the hardware failure contribution to this sequence becomes 1.5(-4) which is not appreciably different than that of IP-2. Thus, we conclude that this sequence should have a frequency on the order of that of the recomputed IP-2 one, namely 2.4(-6)/yr.

3.3.8 Seismic: Loss of Control or AC Power, SE/2RW

The Boolean expression for seismic release category 2RW given on IPPSS report page 7.2-20 was checked and could not be verified.

The expression that we obtained follows:

$$2RW = \overline{32} \vee 15 \vee 18 \vee \Big[(24 \vee 35) \wedge (6 \vee 21 \vee 25 \vee 34) \wedge \Big\{ (6 \vee 27) \wedge \big[(5 \wedge (4 \vee 0.05\ 3)) \vee 23 \vee 26 \vee \big] \Big\} ((7 \vee 33) \wedge (6 \vee 31)) \Big\} \Big].$$

Our understanding is that the IPPSS used the following upper bound expression <u>in</u> the actual calculation.

$$2RW < \overline{32} \wedge (15 \vee 18 \vee \big[19 \wedge (11 \vee 13) \big] \vee 24 \vee 28 \vee 35)$$

We agree that this equation is a reasonable approximation; however, it is not strictly an upper bound.

An integration using the 11 hazard curves from IPPSS report Sections 7.9.1 and 7.9.2 with the 5 fragility curves from IPPSS report Table 7.2-8 was performed using the same relative weighting as the IPPSS and a mean frequency value of 1.6 x 10-6 per year was obtained. This compares to the value of 2.4 x 10-6 reported by the IPPSS. We believe that the difference is due to differences in the integration procedures used and possibly the lumping of hazard curves into the final family used in the DPD operation. A finer discretation of the hazard and fragility points would probably reduce this difference.

The 2RW seismic sequence is the second largest contributor to latent effects. It is dominated by the shear capacity of the control building wall and the capacity of the diesel generator fuel oil tanks, which together have an equivalent median capacity of about 0.8g. We believe that the capacity of the hung ceiling in the control room may be lower and the D&M hazard curves are more representative for the Indian Point site and thus the mean frequency of a 2RW due to seismic effects is 10 times larger, i.e., 2.4×10^{-5}.

We believe that the capacity for the diesel generator fuel oil tanks should be developed based on specific rather than generic data since this component is a major contributor to seismic 2RW.

3.3.9 Tornado and Missiles: Loss of Offsite Power and SW Pumps, SE/2RW

The category 2RW sequence is dominated by the failure of the service water pumps, 11 T, since failure of offsite power will occur at a much lower wind velocity. Loss of Offsite power and the service water pumps leads to a total loss of AC power. Total loss of AC power leads to a seal LOCA and failure of the core cooling systems.

Because the RWST, 9 T, is in series with offsite power, it is not a major contributor to 2RW release. We disagree with the statement in the IPPSS report, page 7.5-19, that the auxiliary feed pump building is a dominant contributor to release category 2RW. This component is not part of the final Boolean expression.

Since missiles from hurricanes are not a significant threat and hurricane wind pressures will not fail the concrete structures, there is no contribution to 2RW from hurricanes. As discussed in review of IPPSS Section 7.5.3, we believe that the failure of the service water pumps due to tornado effects is approximately 10-6 per year. Thus, the mean value of 9.2 x 10-7 per year for category 2RW due to wind loading is reasonable.

3.3.10 Event V: The Interfacing Systems LOCA, V/2

See Section 3.2.15.

3.3.11 Turbine Trip Due to LOP: Loss of All AC, RCP Seal LOCA, Failure to Recover AC Power (Within Three Hours), SE/2RW

This sequence leads to plant state SE and release category 2RW and is the leading internal event with respect to the risk of latent cancers. It differs from the sequence discussed in Section 3.3.7 only in that recovery of AC power is not until after three hours. The difference between three-hour and one-hour recovery has been discussed in Section 3.2.12.

The IPPSS estimated this sequence to have a frequency of 4.8×10^{-7}. The IPPSS estimated the similar sequence at IP 2 to have a frequency of 1×10^{-6}. In Section 3.2.14, we concluded a better estimate should be 5×10^{-7}. In Section 3.3.7, we also concluded that the loss of AC power frequency should be about the same for both IP 2 and IP 3. Since the recovey of AC power is also the same for both plants, we conclude here that the IP 2 frequency estimate for this sequence of 5×10^{-7} also applies to IP 3.

3.3.12 Seismic: Containment Failure, Z-1Q

The sequence leading to release category Z-1Q consists entirely of failure of the containment building shear wall. Because of the relatively high capacity of this failure mode (i.e., median value equal to 1.7g) the mean frequency of failure is only 3.7 x 10-8 per year. This result is sensitive to the upper bound cutoff on the hazard curves. Because we believe the D&M hazard curves are more representative, the IPPSS frequency of Z-1Q is a factor of 2 low.

4.0 Special Issues

4.1 Steam Generator Tube Rupture With Stuck Open Secondary Safety Valve

As discussed in Section 2.2, several errors and omissions were identified in the IPPSS steam generator tube rupture event tree. Because of our several findings, the IPPSS analysis team is performing a revised steam generator tube rupture analysis which will appear as a supplement to the IPPSS.

One of the most potentially risk significant omissions was not modeling a steam generator tube rupture coincident with a stuck open secondary safety valve. If core meltdown occurs, this may be a high risk accident since a direct radioactive material release path to the atmosphere would exist. This type of accident has recently become a concern at NRC because of the Ginna steam generator tube rupture incident which occurred earlier this year. (This was the first U. S. PWR steam generator tube rupture incident in which a secondary safety valve opened.)

We quantified this omission by performing an abbreviated analysis; a simplified event tree was drawn which considered what we felt were potentially significant accident sequences, and event probabilities were estimated based on a review of the Indian Point steam generator tube rupture emergency procedures and our revised system unavailability estimates which appear in Section 2.4. The dominant sequences were identified are presented and discussed below. It can be noted that IP 2's sequence A and B involve the same events as IP 3's sequence A and B. These two sequences will be discussed in turn.

In sequence A, the tube rupture leads to a safety injection signal followed by successful operation of the high pressure injection system (HPIS). The pressure in the secondary of the faulted steam generator will begin to rise and the atmospheric dump valve may be demanded open. The IP 2 emergency procedures instruct the operator to isolate the faulty steam generator and to locally close the dump valve blocking valve. This would eliminate leakage through the dump valve if it failed to close. However, this action may cause the safety valves to be demanded open if the primary system is repressurized above the safety valve set point (e.g., at Ginna this occurred because HPI was not throttled) or if the block valve is closed before the primary pressure is reduced below the safety valve setpoint. (The IP 2 procedures do not give firm guidance as to what the primary pressure must be before closing the block valve and thus we conservatively assume the safety valves will be demanded with a probability of 1.0.) If a secondary safety valve fails to close, the primary system will begin to "steam off" inventory to the atmosphere. We estimate that the high pressure injection

Dominant IP 2 Sequences

$$\text{A)}\ \begin{pmatrix}\text{Steam Generator}\\ \text{Tube Rupture}\\ \text{(.027/R yr)}\end{pmatrix} \cdot \begin{pmatrix}\text{Secondary}\\ \text{Safety Valves}\\ \text{Demanded}\\ \text{Open}\\ \text{(1.0)}\end{pmatrix} \cdot \begin{pmatrix}\text{At Least One}\\ \text{Secondary}\\ \text{Safety Valve}\\ \text{Fails to Close}\\ \text{(.01)}\end{pmatrix} \cdot \begin{pmatrix}\text{Failure}\\ \text{of Residual}\\ \text{Heat Removal}\\ \text{System}\\ (4.4\text{x}10^{-4})\end{pmatrix} = 1.2\text{x}10^{-7}\text{/R yr}$$

$$\text{B)}\ \begin{pmatrix}\text{Steam Generator}\\ \text{Tube Rupture}\\ \text{(.027/R yr)}\end{pmatrix} \cdot \begin{pmatrix}\text{Failure of}\\ \text{High Pressure}\\ \text{Injection System}\\ (5\text{x}10^{-4})\end{pmatrix} \cdot \begin{pmatrix}\text{Secondary}\\ \text{Safety Valves}\\ \text{Demanded Open}\\ \text{(1.0)}\end{pmatrix} \cdot \begin{pmatrix}\text{At Least One}\\ \text{Secondary Safety}\\ \text{Valve Fails}\\ \text{to Close}\\ \text{(.01)}\end{pmatrix} = 1.4\text{x}10^{-7}\text{/R yr}$$

Dominant IP 3 Sequences

$$\text{A)}\ \begin{pmatrix}\text{Steam Generator}\\ \text{Tube Rupture}\\ \text{(.034/R yr)}\end{pmatrix} \cdot \begin{pmatrix}\text{Secondary}\\ \text{Safety Valves}\\ \text{Demanded}\\ \text{Open}\\ \text{(1.0)}\end{pmatrix} \cdot \begin{pmatrix}\text{At Least One}\\ \text{Secondary}\\ \text{Safety Valve}\\ \text{Fails to Close}\\ \text{(.01)}\end{pmatrix} \cdot \begin{pmatrix}\text{Failure}\\ \text{of Residual}\\ \text{Heat Removal}\\ \text{System}\\ (1.9\text{x}10^{-4})\end{pmatrix} = 6.5\text{x}10^{-8}\text{/R yr}$$

$$\text{B)}\ \begin{pmatrix}\text{Steam Generator}\\ \text{Tube Rupture}\\ \text{(.034/R yr)}\end{pmatrix} \cdot \begin{pmatrix}\text{Failure of}\\ \text{High Pressure}\\ \text{Injection System}\\ (4.9\text{x}10^{-4})\end{pmatrix} \cdot \begin{pmatrix}\text{Secondary}\\ \text{Safety Valves}\\ \text{Demanded Open}\\ \text{(1.0)}\end{pmatrix} \cdot \begin{pmatrix}\text{At Least One}\\ \text{Secondary Safety}\\ \text{Valve Fails}\\ \text{to Close}\\ \text{(.01)}\end{pmatrix} = 1.7\text{x}10^{-7}\text{/R yr}$$

system could make up this lost inventory for at least 12 hours. After this time, the refueling water storage tank may empty. To prevent core melt, therefore, the primary system must be depressurized by the AFWS within 12 hours so that the leak rate out the safety can be reduced and so the low pressure residual heat removal (RHR) system can be activated. Once RHR is actuated, the primary system can be brought subcooled and thus steam off to the atmosphere will essentially cease. If RHR fails, steam off will continue and core melt will ensue. The probability of RHR failure we assigned is comprised of only hardware failures and assumes operator errors in establishing RHR are negligible. These actions are very familiar to the operators and are performed every time the plant is brought to cold shutdown.

In sequence B, the tube rupture leads to a safety injection signal followed by failure of the high pressure injection system. As in the previous sequence, we assume the safety valve will be demanded because of the instructions provided in the IP 2 emergency procedures to isolate the atmospheric dump valve. If the valve sticks open, we estimate that within approximately two hours, the core will uncover followed by core meltdown.

Summing these two sequences yields 2.6×10^{-7}/R yr for IP 2 and 2.4×10^{-7}/R yr for IP 3. These sequences would be placed in plant damage state V and release category 2.

As a final note, it should be understood that the sequence frequency estimate for IP 3 was based on IP 2 procedures since we did not have a current version of the IP 3 procedure.

4.2 Core Melt/Systems Interactions

As mentioned in Section 2.2.1, the Indian Point event trees imply that the containment spray system and containment fan cooler system may be utilized to scrub radioactivity and protect the containment from overpressure during a core melt accident. The fault tree analysis of these systems also assume that the system reliability will not be degraded due to the adverse environment within containment following a core melt. In this section we will investigate the affect that not giving credit for these systems has on the IPPSS release category estimates.

Results from the Sandia experiments and discussions with pump experts at Babcock and Wilcox suggest that spray recirculation system pumps may fail during a core melt accident. The experiments indicate that during the core meltdown process, millions of solidified metal droplets of various sizes would be ejected when the molten core interacts with the concrete in the cavity below the reactor vessel. Following a core meltdown, it is reasonable to assume that the water in the containment sump would be contaminated with these metal chips. Pump experts at Babcock and Wilcox feel that containment spray pumps may sieze if the sump water contains small metal chips. This proposed core melt/spray pump interaction does not affect the IPPSS plant damage state or release category frequencies since, as stated in Section 2.2.1, the analysis does not give credit for operation of the spray recirculation system.

Following a core meltdown, the fan cooler system may possibly fail by one or a combination of the following mechanisms:

1) cabling or instrumentation failure due to containment hydrogen burns,

2) cabling or instrumentation failure due to radiation exposure, or

3) plugging of fan cooler filters or cooling coils due to aerosol generation.

The IPPSS analysis team do not feel these are likely failure mechanisms for the following reasons:

1) most and possibly all important fan cooler cabling either are adequately shielded from the containment atmosphere or the insulation exhibits combustion retardant properties,

2) the cabling should handle the radiation doses expected to exist at the location of the cables following the melt, and

3) the amount of aerosols reaching the coolers should be insignificant since most small aerosols (2 to 4 micron) will be scrubbed out in the water in the reactor cavity and larger aerosols (100 micron--1 mm) will fall out due to gravity before reaching the fans.

Due to the limited time available to perform this review, and the fact that these issues are currently being addressed in several NRC and Sandia equipment qualification research programs, we did not attempt to resolve these issues. Rather, a sensitivity analysis was performed which investigated the effect that assuming fan cooler failure has on the IPPSS release categories.

The IPPSS mapping from plant damage states to significant release categories can be approximated by the following relations:

Release Categories		Plant Damage States
Z-1Q	=	Direct Containment Failure Due to Seismic$_E$
Z-1	=	$AE_I + AL_I + AE_E$
2	=	V_I
2RW	=	$SE_I + SL_I + TE_I + SE_E + TE_E$
8A	=	$SEF_I + SLF_I + TEF_I + AEF_I + ALF_I + SEF_E + TEF_E + AEF_E$
8B	=	$SEFC_I + SEC_I + SLFC_I + SLC_I + TEFC_I + TEC_I + AEFC_I + AEC_I + ALFC_I + ALC_I\ SEFC_E + SEC_E + TEFC_E + TEC_E + AEFC_E + AEC_E$.

The subscripts I and E refer to internal and external plant damage states respectively.

If one assumes the fan coolers will fail during a core melt and the containment spray injection system is not available during the recirculation phase (see discussion in Section 2.2.1), the mapping from IPPSS plant damage states to release categories could be approximated by the following:

Release Categories		Plant Damage States
Z-1Q	=	Direct Containment Failure
Z-1	=	$AE_I + AL_I + AE_E + AEF_I + ALC_I + AEF_E$
2	=	V
2RW	=	$SE_I + SL_I + TE_I + SE_E + TE_E + SEF_I + SLF_I + TEF_I + SEF_E + TEF_E + SLFC_I + SLC_I + ALFC^*_I + ALF^*_I$

$$8A = 0$$

$$8B = SEFC_I + SEC_I + TEFC_I + TEC_I + AEFC_I + AEC_I + SEFC_E + SEC_E + TEFC_E + TEC_E + AEFC_E + AEC_E.$$

The damage states with an asterisk were mapped with the aid of Brookhaven National Laboratories.

The IPPSS damage state frequencies can be obtained from Table 8.5.3-1/Table 8.5.3-18 for internal events, and deduced by comparing Table 8.3-9/Table 8.3-10 with analysis presented in IPPSS Section 7 for external events. Table 4.2-1 compares the IPPSS release category frequency estimates with an alternate set of estimates which assumes the fan coolers fail following a core melt and the spray injection system is not available during the recirculation phase.

The most significant increases in the severe release category frequencies and thus the risk, appears in category 2RW for Indian Point 3. However, since this total is a factor of two, less than the IPPSS 2RW frequency for Indian Point 2, the overall risk at the Indian Point site should not be appreciably affected.

TABLE 4.2-1

Release Category	IPPSS Indian Point 2 Release Category Frequencies	Alternate Indian Point 2 Release Category Frequencies	IPPSS Indian Point 3 Release Category Frequencies	Alternate Indian Point 3 Release Category Frequencies
Z-1Q	6.8×10^{-7}	6.8×10^{-7}	3.7×10^{-8}	3.7×10^{-8}
Z-1	1.7×10^{-8}	1.7×10^{-8}	4.7×10^{-9}	5.9×10^{-9}
2	5.4×10^{-7}	5.4×10^{-7}	4.9×10^{-7}	4.9×10^{-7}
2RW	3.[illegible] $\times 10^{-4}$	4×10^{-4}	6.5×10^{-5}	1.7×10^{-4}
8A	4.8×10^{-5}	0	1.1×10^{-6}	0
8B	9.1×10^{-5}	5.6×10^{-5}	1.3×10^{-4}	8.5×10^{-5}

4.3 Feed and Bleed Capability

The IPPSS gave credit for post shutdown decay heat removal via feed and bleed (FB) core cooling. FB would be utilized during small LOCAs and transients if the auxiliary feedwater system (i.e., the normal decay heat removal system) was unavailable. Iniciation of FB at Indian Point requires the operator to:

a. Recognize that auxiliary feedwater and secondary heat removal has failed.

b. Start a safety injection pump (if pressure is low enough).

c. Open both pressurizer power operated relief valves and their associated block valves.

d. Verify that adequate heat removal is taking place.

FB is currently not a fully accepted core cooling method at the NRC. We have been asked to assess the affect that giving credit for FB has on the core melt frequency and on the risk calculated in the IPPSS. Before presenting the quantitative results, a discussion of the Indian Point operator training and emergency procedures regarding FB, and the IPPSS modeling of FB is in order.

Discussions with plant operators at both Indian Point units revealed that they received FB simulator training. However, a review of Indian Point emergency procedures revealed that no FB procedures exist at Unit 2 and FB procedures are available at Unit 3 in response to small LOCAs only. The IPPSS has therefore made some assumptions regarding FB operator actions which are not supported by plant emergency procedures. The IPPSS assigned a probability of 3.9×10^{-4} that the operator would fail to establish feed and bleed. We feel this probability is optimistic and would suggest a probability closer to 0.1. (In other PRAs with which we are familiar, 0.1 is typically assigned to accident situations in which no or inadequate emergency procedures exist but the postulated operator action seems likely.)

If it is assumed that feed and bleed cooling is not possible, one replaces the IPPSS probabilities quoted for event tree events OP-1, OP-2, and OP-5 with 1.0. This was done for the dominant accident sequences for each event tree and includes the affect of other significant findings of this letter report. The "no-feed and bleed" dominant accident sequences are summarized in Table 4.3-1. As can be seen from the table, assuming feed and bleed is not possible primarily affects plant damage state TEFC.

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. We do disagree, however, with the failure probability the IPPSS assigned to feed and bleed. As mentioned above, we feel the probability should be closer to 0.1. This IPPSS nonconservatism is somewhat offset by the IPPSS conservative assumption that main feedwater is always unavailable following an initiating event. Discussions with Indian Point personnel indicate that following most initiating events, main feedwater remain in operation at decay heat flow rates. Data appearing in the ANO PRA indicates that main feedwater remains in operation approximately 94 percent of the time following initiating events caused by reactor trips and turbine trips. Data appearing in NUREG/CR-2497 "Precursors to Potential Severe Core Damage Accidents" indicates that approximately 50 percent of loss of main feedwater initiating events at Westinghouse PWRs are recovered within the short term. As our best estimate, then, we modify the sequences presented in Table 4.3-1 with our probability estimates of feed and bleed core cooling and main feedwater operation. The results of this exercise are presented in Table 4.3-2.

Table 4.3-1

No Feed and Bleed Dominant Sequences

Event Tree Sequence	Indian Point 2	Indian Point 3
Loss of Main Feedwater Sequence 9	2×10^{-4}	1.1×10^{-4}
Closure of One MSIV Sequence 9	3.8×10^{-5}	2.7×10^{-6}
Loss of RCS Flow Sequence 9	4.1×10^{-6}	5.1×10^{-6}
Turbine Trip Sequence 9	2.2×10^{-4}	8.1×10^{-5}
Loss of Offsite Power Sequence 9	6.7×10^{-6}	9×10^{-6}
Reactor Trip Sequence 9	2×10^{-4}	8.7×10^{-5}
ATWS (See Section 4.4.1 of this report)	6.7×10^{-5}	7.4×10^{-5}

Table 4.3-2

Revised Feed and Bleed Dominant Sequences

Event Tree Sequence	Indian Point 2	Indian Point 3
Loss of Main Feedwater Sequence 9	$1x10^{-5}$	$5.5x10^{-6}$
Closure of One MSIV Sequence 9	$1.9x10^{-6}$	$1.4x10^{-7}$
Turbine Trip Sequence 9	$1.3x10^{-6}$	$4.9x10^{-7}$
Loss of Offsite Power Sequence 9	$5.4x10^{-7}$	$8.1x10^{-7}$
Reactor Trip Sequence 9	$1.2x10^{-6}$	$5.2x10^{-7}$
ATWS (See Section 4.4.1 of this report)	$6.7x10^{-6}$	$7.4x10^{-6}$

4.4 Proposed Indian Point Plant Design Modifications as a Result of the IPPSS

The IPPSS was not totally based on the current design of the Indian Point plants. During the course of the analysis, five potential problem areas relating to plant design were identified. Consolidated Edison and PASNY recognized these problems and committed to implement modifications to correct them. The IPPSS was based on the future Indian Point plant designs after the modifications are installed. The five modifications are listed below:

1) System modification, procedural change, or verification testing to ensure that sufficient back-pressure will be maintained in the service water system to prevent service water pump overload for cases when only one service water pump is operating with the system in accident configuration.

2) Rearrangement of diesel generator fuel oil transfer pump power supplies such that the primary transfer pump for each diesel is powered from one of that diesel's electrical buses. (Indian Point Unit 2 only)

3) Replacement of manual isolation valves with motoroperated isolation valves in certain of the fan cooler service water discharge lines. (Indian Point Unit 2 only)

4) Implementation of masonry wall upgrading modifications for station batteries in response to IE Bulletin SO-11.

5) Implementation of plant modifications for mitigation of ATWS.

During the review of the IPPSS, it was learned that PASNY and ConEd changed their minds about implementing the ATWS modification. The IPPSS analysis of ATWS events therefore needed revision. The revised ATWS analysis appears in Section 4.4.1. Our review verified that plants are still intending to implement the other four modifications. Three of the four modifications are fairly straight forward; this allowed us to review the proposed modifications to ensure design adequacy. However, the proposed modification dealing with the service water system was not as clear. We did not feel comfortable in our review of this modification, since inadequate information was available describing it. In Section 4.4.2 we investigate the affect that not implementing the service water system modification would have on the IPPSS results.

4.4.1 ATWS Modification

The IPPSS analysis was based upon an ATWS modification which would make turbine trip independent of reactor trip. Nonimplementation of this modification results in a much higher peak RCS pressure following a loss of main feedwater ATWS event than was modeled to occur in the IPPSS.

In response to a transient initiated by a loss of main feedwater, a trip signal is sent to the reactor. Upon opening of the reactor trip breakers, a trip signal is sent to the main turbine. Due to this series relationship, failure of reactor trip will cause failure of turbine trip. NUREG-0460 indicates that at Westinghouse plants a loss of main feedwater followed by reactor trip and turbine trip failure can result in a peak RCS pressure of 3800 psi or greater.

The IPPSS analysis of ATWS events assumed that pressures exceeding 3200 psi cause failure of core cooling systems and thus lead to core melt. Other PRAs (e.g., ANO, Crystal River) did not make this assumption and thus we feel the 3200 psi criteria is conservative. As mentioned in Section 2.2.2 of this report, IPPSS personnel are performing a new ATWS analysis which will appear as a supplement to the IPPSS. Discussions with personnel performing the reanalysis revealed that the 3200 psi core melt criteria will not be adopted in the new version. They sited material presented in a Westinghouse document, Appendix C of WCAP-8330, which indicates that a 3800 psi pressure spike has a very small probability (subjectively estimated to be no greater than 1×10^{-2}) of leading to core cooling system failure. (System failure is postulated to occur via destruction of the high pressure injection check valves.) The document also indicates that a small LOCA is likely to occur as a result of the pressure spike.

The preliminary quantitative results of the IPPSS reanalysis were discussed with us. We checked their calculations and compared their analysis with other PRA ATWS analyses. The frequency and contributors to the dominant sequences we have identified are listed below:

A) (Loss of main feed) · (Power level above 50 percent) · (failure to scram) · (failure of turbine trip) · (failure of high pressure injection due to pressure spike)

B) (Loss of main feed) · (power level above 50 percent) · (failure to scram) · (failure of turbine trip) · (failure of feed and bleed core cooling)

<u>Indian Point 2</u>

$$A = (6.7)\ (.5)\ (2 \times 10^{-5})\ (1.0)\ (10^{-2}) = 6.7 \times 10^{-7}$$
$$B = (6.7)\ (.5)\ (2 \times 10^{-5})\ (1.0)\ (.1) = 6.7 \times 10^{-6}$$

<u>Indian Point 3</u>

$$A = (3.8)\ (.5)\ (3.9 \times 10^{-5})\ (1.0)\ (10^{-2}) = 7.4 \times 10^{-7}$$
$$B = (3.8)\ (.5)\ (3.9 \times 10^{-5})\ (1.0)\ (.1) = 7.4 \times 10^{-6}$$

We therefore estimate the ATWS core melt frequency for Indian Point 2 to be $\sim 7.4 \times 10^{-6}$, and for Indian Point 3 $\sim 8.1 \times 10^{-6}$. These sequences would result in plant damage state SEFC. The IPPSS reported values for ATWS were $\sim 1.3 \times 10^{-6}$ for Indian Point 2 and $\sim 1.1 \times 10^{-6}$ for Indian Point 3.

It should be noted that the 0.1 probability we assigned to failure of feed and bleed core cooling is significantly greater than the IPPSS reanalysis values (6.1×10^{-3} and 8.9×10^{-3} for Indian Point 2 and 3, respectively). The .1 value is dominated by failure of the operators to perform the actions necessary to establish feed and bleed. Since the Indian Point plants have no ATWS emergency procedures, very limited or no feed and bleed procedures, and due to the high stress situation, we feel a probability of .1 is more realistic. This value was also assigned to a similar situation in the ANO PRA.

4.4.2 Service Water System Modification

The nuclear header portion of the Indian Point service water system provides cooling to the five containment fan coolers and the three diesel generators. In response to an ES signal, two of three nuclear header service water pumps must operate to provide adequate cooling to these eight components. Prior to an ES signal, one of three pumps is required since the service water delivery to fan coolers is reduced substantially. This is because two service water air operated valves at the fan cooler discharge remain closed. However, during the IPPSS analysis of loss of offsite power sequences, it was realized that two of three service water pumps may be needed well before an ES signal is generated.

Following a loss of offsite power (LOP), valves have a good chance of opening due to a loss of instrument air. The instrument air compressors must be reloaded manually by the operators and the valves could open prior to the performance of this action. Also, if a diesel fails, the opertor may not choose to reload instrument air because of other competing diesel generator loads. If these valves open, two of three service water pumps would be required to cool the diesel generators. Since each of the three diesels powers one of the three service water pumps, this implies that if two diesels fail to start, the third will also fail due to a lack of service water. The IPPSS analysis assumes a plant modification will be installed which will prevent the fan cooler discharge valves from opening following a LOP. This modification would conceivably prevent failure of the third diesel, given failure of the first two.

The calculations presented below assumes that the modification is not installed, and thus, failure of two diesels leads to failure of all three. The dominant Indian Point LOP sequences would be approximated as:

$$(.18)(3)(1.5 \times 10^{-3})(.05) = 4 \times 10^{-5}, \text{ category 8B}$$
$$(.18)(3)(1.5 \times 10^{-3})(.05)(.14) = 5.8 \times 10^{-6}, \text{ category 2RW}$$

where

.18 = frequency of loss of offsite power,
3 = number of combinations of two failed diesels
1.5×10^{-3} = probability of failing two diesel generators (Q_o from Section 3.2.4)
.05 = probability of failing to restore offsite power or the gas turbines within one hour (from Section 3.2.4)
.17 = probability of failing to restore offsite power or the gas turbines within one to three hours (from Section 3.2.12).

It should be noted that recovery of diesel generators is not given credit because of the complications of reestablishing service water to them.

The following changes would be made to the dominant accident sequence tables (Table 8.3-9 for Indian Point 2, Table 8.3-10 for Indian Point 3).

Table 8.3-9

a) Replace category 8B sequences 4, 10, 12 with the category 8B sequence calculated above. This does not significantly change the frequency estimate of LOP sequences which lead to 8B.

b) Replace category 2RW sequence 16 with the category 2RW sequence calculated above. This would increase the frequency estimate of LOP 2RW sequences by a factor of 6.

Table 8.3-10

a) Replace category 8B sequence 7 with the category 8B sequence calculated above. This would increase the frequency estimate of LOP 8B sequences by a factor of 15.

b) Replace category 2RW sequence 16 with the category 2RW sequence calculated above. This would increase the frequency estimate of LOP 2RW sequences by a factor of 12.

4.5 Reactor Coolant Pump Seal LOCA

Several of the IPPSS dominant internal and external accident sequences involve reactor coolant pump (RCP) seal failure. Seal failure is assumed to occur following failure of the redundant means of providing seal cooling (i.e., charging system and component cooling system) and is predicted to lead to a 1200 gpm LOCA at 30 minutes. The reason that seal LOCAs appear in so many dominant sequences is because failure of AC power causes common mode failure of the seal cooling systems and the emergency core cooling safety injection pumps.

If however, a seal LOCA did not occur following loss of seal cooling, the reactor coolant system would not lose inventory and the safety injection pumps would not be required. With an intact reactor coolant system, decay heat could be removed with the AC independent turbine driven auxiliary feedwater pump via the steam generators. In this section, we assume that a seal LOCA will not occur following a loss of seal cooling and requantify the Indian Point dominant accident sequences.

We suspect that the seal LOCA may not occur for two reasons. One, the Westinghouse memorandum upon which the IPPSS 1200 gpm assumption was based is a very simplistic bounding analysis. Two, an experiment performed on a Byron Jackson RCP showed that significant leakage did not occur for 56 hours following interruption of seal cooling to a static RCP seal. (Memorandum from J. Zudans, NRC, to Z. Rosytoczy, NRC, Subject: St. Lucie 2; Reactor Coolant Pump Seal Hot Standby Test, September 19, 1980.) We recognize that Byron Jackson RCP seals are not identical to Westinghouse RCP seals. However, similarities do exist which might indicate that Westinghouse seals would not leak significantly.

IPPSS personnel were requested to identify the dominant internal and external accident sequences under the assumption that a seal LOCA does not occur following a loss of seal cooling. The sequences they identified were reviewed and revised by us, where necessary, to reflect our other findings delineated in this report. The results of this exercise are listed below in terms of dominant risk significant (2RW release category) accident sequences and dominant core melt (8A, 8B release category) accident sequences.

The value of 0.1 appearing in the "Notes" column of Table 4.5-1 represents failure of the operator to take local control of the AFWS turbine pump following a total loss of AC power or during certain postulated fire scenarios. Following a total loss of AC power, the turbine pump must be controlled locally

Table 4.5-1

2RW Release Category - Indian Point 2

Sequence	Frequency	Notes
• Hurricane: Loss of Control or Power	5.4×10^{-4}	a) Frequency estimate taken from Section 3.2.5 b) Unaffected by seal LOCA since dominated by control room failure which also fails AFWS turbine pump control
• Seismic: Loss of Control or Power	2.8×10^{-4}	a) Frequency taken from Section 3.2.1. b) Unaffected by seal LOCA since dominated by control room failure which also fails AFWS turbine pump control
• Tornado and Missiles: Loss of Control Power	1.6×10^{-5}	a) Frequency taken from IPPSS Table 8.3-9 b) Unaffected by seal LOCA since dominated by control room failure which also fails AFWS turbine pump control
• Fire: Electrical Tunnel (CB End)	1.2×10^{-5}	a) $1.2 \times 10^{-5} = (1.2 \times 10^{-4})(.1)$, where 1.2×10^{-4} taken from Section 2.7.4 of this report and 0.1 is the probability of failing to take local control of the AFWS turbine pump.
• Fire: Electrical Tunnel (PAB End)	1×10^{-5}	a) $1 \times 10^{-5} = (1 \times 10^{-4})(.1)$, where 1×10^{-4} taken from Section 2.7.4 of this report and 0.1 is the probability of failing to take local control of the AFWS turbine pump.
• Fire: Cable Spreading Room	2.3×10^{-6}	a) Frequency taken from Section 2.7.4. b) Turbine pump failure already factored into this estimate
• Turbine Trip Due to Loss of Offsite Power: Failure of All AC Power and Turbine Driven AFWS Pump and Failure to Restore AC Within 3 Hours	5×10^{-8}	a) $5 \times 10^{-8} = (5 \times 10^{-7})(.1)$, where 5×10^{-7} taken from Section 3.2.14 and 0.1 is the probability of failing to take local control of the AFWS turbine trip
Total	8.6×10^{-4}	

Table 4.5-1 (Cont.)

8A/8B Release Category - Indian Point 2

Sequence	Frequency	Notes
Large LOCA: Failure of Recirculation Cooling	3.9×10^{-5}	a) Frequency taken from Section 3.2.8 b) Unaffected by seal LOCA
Medium LOCA: Failure of Recirculation Cooling	3.9×10^{-5}	a) Frequency taken from Section 3.2.9 b) Unaffected by seal LOCA
Fire: Cable Spreading Room	1.2×10^{-5}	a) Frequency taken from 2.7.4 b) Turbine pump failure already factored into this estimate
Small LOCA: Failure of Recirculation Cooling	1.2×10^{-5}	a) Frequency taken from Section 3.2.7 b) Unaffected by seal LOCA
Loss of Main Feedwater: Failure of AFWS, Feed and Bleed Cooling, and Failure of Main Feedwater Recovery	1×10^{-5}	a) Frequency taken from Section 4.3 b) Unaffected by seal LOCA
Small LOCA: Failure of High Pressure Injection	9.3×10^{-6}	a) Frequency taken from Section 3.2.13 b) Unaffected by seal LOCA
(others)		
Turbine Trip Due to Loss of Offsite Power: Failure of All AC Power and Turbine Driven AFWS Pump. AC Restored Between 1-3 hours	2.4×10^{-7}	a) $2.4\times10^{-7} = (2.4\times10^{-6})(.1)$, where 2.4×10^{-6} taken from Section 3.2.10 and 0.1 is the probability of failing to take local control of the AFWS turbine pump
Total	$\sim1.2\times10^{-4}$	

Table 4.5-1 (Cont.)

2RW Release Category - Indian Point 3

Sequence	Frequency	Notes
Seismic: Loss of Control	2.4×10^{-5}	a) Frequency taken from Section 3.3.8 b) Unaffected by seal LOCA since dominated by control room ceiling failure which also fails AFWS turbine pump control
Fire: Switchgear Room	1.4×10^{-5}	a) $1.4 \times 10^{-5} = (1.4 \times 10^{-4}(.1)$, where 1.4×10^{-4} taken from Section 2.7.4 and 0.1 is the probability of failing to take local control of the AFWS turbine pump
Fire: Cable Spreading Room (Tunnel Entrance)	7.2×10^{-6}	a) $7.2 \times 10^{-6} = (7.2 \times 10^{-5})(.1)$, where 7.2×10^{-5} taken from Section 2.7.4 and 0.1 is the probability of failing to take local control of AFWS turbine pump
Fire: Cable Spreading Room (North Wall)	3.3×10^{-6}	a) Frequency taken from 2.7.4 b) Turbine pump failure already factored into this estimate
Fire: Upper Cable Tunnel	3×10^{-6}	a) Frequency taken from 2.7.4 b) Turbine pump failure already factored into this estimate
Fire: Switchgear Room	2.7×10^{-6}	a) Frequency taken from 2.7.4 b) Turbine pump failure already factored into this estimate
Turbine Trip Due to Loss of Offsite Power: Failure of All AC Power and Turbine Driven AFWS Pump and Failure to Restore AC Within 3 Hours	5×10^{-8}	a) $5 \times 10^{-8} = (5 \times 10^{-7})(.1)$, where 5×10^{-7} taken from Section 3.3.11 and 0.1 is the probability of failing to take local control of the AFWS turbine pump
Total	5.4×10^{-5}	

Table 4.5-1 (Cont.)

8A/8B2 Release Category - Indian Point 3

Sequence	Frequency	Notes
Large LOCA: Failure of Low Pressure Recirculation Cooling	3.9×10^{-5}	a) Frequency taken from 3.3.3. b) Unaffected by seal LOCA
Medium LOCA: Failure of Low Pressure Recirculation Cooling	3.9×10^{-5}	a) Frequency taken from 3.3.4 b) Unaffected by seal LOCA
Fire: Cable Spreading Room (North Wall)	1.8×10^{-5}	a) Frequency taken from 2.7.4 b) Turbine pump failure already factored into this estimate
Small LOCA: Failure of High Pressure Recirculation Cooling	1.5×10^{-5}	a) Frequency taken from 3.3.1 b) Unaffected by seal LOCA
Small LOCA: Failure of Safety Injection	9.8×10^{-6}	a) Frequency taken from 3.3.6 b) Unaffected by seal LOCA
Turbine Trip Due to Loss of Offiste Power: Failure of All AC Power and Turbine Driven AFWS Pump. AC Restored Between 1-3 Hours	2.4×10^{-7}	a) $2.4 \times 10^{-7} = (2.4 \times 10^{-6})(.1)$, where 2.4×10^{-6} taken from 3.3.7 and 0.1 is the probability of failing to take local control of the AFWS turbine pump
Total	1.2×10^{-4}	

because instrument air is lost. Plant personnel indicate this must be done within approximately one hour. During certain postulated fire scenarios, as discussed in 2.7.4, the operator must control the pump locally because instrumentation is lost in the control room. During the fire, the operator relies on instrumentation outside the control room and which is a considerable distance from the AFWS pump room. We have assigned a probability of failure of .1 to both of these situations because, as stated previously, in Sandia PRAs as a first cut, we typically assign this value for accident situations in which no emergency procedures exist but the postulated operator action seems likely.

4.6 Loss of Component Cooling Water Due to a Pipe Break

As stated in Section 2.1, the IPPSS did not analyze this initiating event. This section will assess the impact that this omission has on the IPPSS results.

If a pipe break occurs in one of several of the larger component cooling water system lines, the 25000 gallon system would empty in a short time (e.g., approximately 5 minutes). Loss of this water means that the following important equipment will not receive cooling:

a) four reactor coolant pump (RCP) thermal barrier heat exchangers,

b) three charging pump oil coolers, and

c) three safety injection pump oil coolers.

The IPPSS predicts that failure of a) and b) will lead to a 1200 gpm RCP seal LOCA within 30 minutes. Indian Point plant personnel predict that each charging pump will operate 5 minutes without cooling. Since the charging pumps would be operated in succession, the seal LOCA would occur 30 minutes after failure of these pumps, or approximately 50 minutes after the pipe break (5 minutes to empty system, 15 minutes to fail charging pumps and 30 minutes for seal LOCA).

Following the seal LOCA, all three safety injection pumps will actuate automatically upon low RCS pressure. Indian Point personnel predict that each safety injection pump will operate 5 minutes without cooling. Since these pumps must operate to prevent core melt, a core melt accident will be assured unless cooling to the safety injection pumps is not restored in about 1 hour following the pipe break (50 minutes from pipe break initiation to seal LOCA, 5 minutes to fail all three safety injection pumps following the seal LOCA).

At Indian Point 2, the following two operator actions could be performed to recover from this accident:

1) realign manual valves and establish city water cooling to the charging pumps within approximately 20 minutes following the pipe break, or

2) realign manual valves and establish city water cooling to the safety injection pumps within approximately 1 hour.

We assign a probability of .5 of failing to perform action 1) since the IP 2 loss of component cooling water procedure instructs (though not very explicitly) the operator to perform this action but the time available is small. We assign a probability of 0.5 of failing to perform action 2) since the procedures do not address the realignment of city water to the injection pumps early in the accident but discussions with the operators revealed they were aware of the city water connection. The 0.5 value also reflects a somewhat larger time available to perform the action. The total nonrecovery probability for IP 2 is therefore .25.

At Indian Point 3, the following operator action could be performed to recover from this accident:

1) connect a spool piece and establish city water cooling to the charging pumps within approximately 20 minutes

We assign a probability of 0.9 of failing to perform this action since the IP 3 loss of component cooling water procedure does not address this action but discussion with the operators revealed that they were aware of the city water connection. The .9 value also reflects the fact that connection of a spool piece is unlikely within the 20-minute time window. (It should be noted that no safety injection pump city water connection exists at IP 3.)

The total sequence frequency for IP 2 and IP 3 is calculated as:

<u>IP 2</u>

$$(1.5\times10^{-4}\ \text{LOCAs/R yr})\cdot(.25) = 3.8\times10^{-5}/\text{R yr},$$

<u>IP 3</u>

$$(1.5\times10^{-4}\ \text{LOCAs/R yr})\cdot(0.9) = 1.4\times10^{-4}/\text{R yr}$$

These sequences would result in plant damage state SEFC and release category 8B. In these calculations, we derived the loss of component cooling water pipe break frequency from the piping analysis of this system presented in the IPPSS (see pages 1. -800, 801, 1.6-778, and 779). We feel this value is a reasonable estimate since it compares favorably with the large LOCA frequency presented in WASH-1400, i.e., 1×10^{-4}. It should be noted here, however, that quantification of pipe breaks involves large uncertainties and is generally believed to be a "black art." Because of this fact, we would suggest that the plants implement modifications to improve the probability of recovering from the pipe break.

4.7 Completeness

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.

- Pressurized thermal shock--discussed in Section 2.1 and not evaluated in this review.
- Steam generator tube rupture coincident with a stuck open secondary safety valve--discussed and evaluated in Section 4.1.
- Hot gas layer failure mode of safety system cabeling during a fire--discussed and evaluated in Section 2.7.4.
- The Ramapo Fault was not considered as a source zone in the seismic analysis--discussed and evaluated in Section 2.7.1.
- Safety System failure caused by core meltdown phenomena--discussed and evaluated in Section 4.2.
- An initiating event caused by a pipe break in the component cooling water system--discussed and evaluated in Section 4.6.
- Wind channelization of hurricane winds--discussed and evaluated in Section 2.7.2
- Low pressure system and containment spray system ß factors were omitted--discussed and evaluated in Section 2.4.
- Reactor coolant pump seal ruptures were not included in the small LOCA initiating event data base--discussed and evaluated in Section 2.1.
- Steam generator overfill scenarios were not considered--not discussed or evaluated in this report.
- Unit 3 control room ceiling failure due to a seismic event--discussed and evaluated in Section 3.3.8.
- Ground roughness and shape of building effects on wind dispersion--discussed in Section 2.7.2.

5. Summary and Conclusions

Over the past three and one-half months, we have reviewed the Indian Point Probabilistic Safety Study. Our review, at this time, was limited to the treatment of the plant systems and external events. This section summarizes some of our more substantive findings.

Section 5.1 lists several of the more important findings in Sandia's review of the Indian Point Probabilistic Safety Study (IPPSS). Section 5.2 presents our recommended estimate of plant damage state frequencies for use in the containment and consequence analysis. This estimte reflects, to the degree possible given the limited scope of our reveiw, our best current judgement of these frequencies. Included in these estimates are the significant quantitative conclusions presented in the text. Section 5.2.1 summarizes our findings for the internal events, and Section 5.2.2 summarizes our findings for the external events. Section 5.2.3 combines these, and Section 5.2.4 highlights the sensitivity issues investigated.

In general, we found that the systems analysis portion of the study to be consistent in scope and detail with ongoing probabilistic risk assessments. The treatment of external events represents an advancement over what has been done in the past. We commend the IPPSS analysis team for their utilization of plant-specific data in their analysis.

Unfortunately, we found the documentation for the report, though voluminous, often lacking. This made review difficult and, at times, raised questions. Many of these questions, however, were resolved through the cooperation of those who performed the study.

Our principal findings are summarized in the following section. By the very nature of the review process, we concentrate on negative findings and impressions with respect to the IPPSS. We have tried, however, to place these in perspective with respect to their impact on the frequency of core melt and risk. In some instances, we note where the Indian Point treatment appears reasonable to us.

5.1 Important Findings

Among the important findings of our review are the following, grouped by topic:

<u>Initiating Events</u>

- 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.

- 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.

- The intiating event frequencies for each plant are based on the operating history of each plant.

Event Trees

- 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.

- 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.

- 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.

- Feed and bleed capability is given more credit than the procedures indicate it should.

- 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.

Success Criteria

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

Fault Trees

- 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.

- 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.

- In the degraded power states, the IPPSS often ignored maintenance unavailability for the pumps which could still receive power.

- In the sequence evaluations for the loss-of-offsite power initiating event, the wrong service water system unavailability was used.

- 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.

- Several errors were identified in the analysis of the auxiliary feedwater system. However, their effect was shown to be of small importance.

- 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.

Human Reliability Analysis

- 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;

  - personal estimates of operator performance rather than using simple measurements;
  - inadequate documentation of the use of expert opinion;

  - optimistic assessments of dependence among tasks done by the same person;

  - apparent nonconsideration of some possibilities for common cause failures from human errors;

- possible insufficient consideration of errors in restoring safety components after test, maintenance, or calibration.

o 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.

Estimation Methodology

o Indian Point's estimates of maintenance unavailabilities appear to be consistent with Indian Point data.

o 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 V analysis used the 5 and 95% bounds.)

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

o The inclusion of the β-factor for accounting for "other", dependent causes of failure is inconsistent.

External Events

o For the seismic hazard and fragility analysis, the methodologies used in the IPPSS are appropriate and adequate to perform a seismic risk analysis.

o The Ramapo fault zone was not included in the analysis and should be addressed.

o In general, the uncertainty of the parameters used in the seismic analysis are understated, but the median values are considered to be conservative.

o 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.

- The tornado hazard curves are conservative, but the hurricane hazard curves are nonconservative.

- Many statements in the wind fragility analysis are undocumented.

- 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.

- The conversion of pressure to equivalent wind velocity ignores the shape factors of the buildings.

- The analysis presented in IPPSS for the loss-of-offsite power caused by wind is unconservative.

- 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.

- 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.

- The systems/components considered in the seismic and wind logic models seem to be reasonably complete.

- 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.

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

- 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.

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

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

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

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

o 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 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.

o 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.

o 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.

Accident Sequence Analysis

o In general, the IPPSS accident sequence analysis was difficult to follow because of

- incorrect and/or incomplete references.

- nonmatching numerical results.

- unclear or inadequate description of events or the modeling of them.

o 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.

o 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.

- The assumption that loss of power to busses, 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.

- 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.

- Tornado initiated sequences at IP-2 appear to have been reasonably estimated.

- 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.

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

- 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.

- 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 re-analyzed. The results of the revised analysis are not appreciably different than those presented in the IPPSS.

- 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.

- In the IPPSS, some IP-3 sequence frequencies are higher than those of identical IP-2 sequences from what seems an over-application of the data. We feel this is not justified.

5.2 Estimated Plant Damage State/Release Category Frequencies and Sensitivity Issues

5.2.1 Internal Events

Tables 5.2-1 and 5.2-2 summarizes the effect that the findings discussed in the previous sections have on the Indian Point Unit 2 and 3 internal event plant damage states and release category frequencies.

The first column is a listing of 21 plant damage states defined in the IPPSS. The nomenclature is: S or A denotes small or large LOCA, T denotes transient, V denotes interfacing systems LOCA, E or L denotes early or late core melt, F and C denotes fans and sprays working, respectively. Also appearing in column one are the frequencies of those damage states as calculated in the IPPSS.

The second column represents the IPPSS significant internal event release categories. The nomenclature is: 8B denotes a core melt in which the spray system operates and the containment does not fail followed by a release at the design basis leak rate, 8A denotes a core melt in which the spray system fails and the containment does not fail followed by a release at the design basis leak rate, 2RW denotes a core melt followed by a late overpressure failure of containment, 2 denotes a core melt followed by a large containment bypass failure, and Z-1 denotes a core melt followed by an early overpressure failure of containment. Also appearing in column two are the frequencies of these release categories as calculated in the IPPSS. These frequencies are aproximately the summation of the plant damage states of the box to the left, e.g., SE+SL+TE = 2RW.

The third and fourth columns represent the revised estimates of the plant damage state and release category frequencies based on the significant findings in Sections 2 through 4. It can be noted that a dash appears instead of a frequency estimate in several places. A dash denotes that we did not attempt to recalculate a frequency because these damage states and release categories were found to have a small impact on risk as calculated in the IPPSS.

<u>Unit 2 Internal Events - Table 5.2-1</u>

Via comparison, it can be noted that 8 of the 21 IPPSS damage state frequencies have been revised for Unit 2. These revisions are summarized below.

Table 5.2-1. Indian Point 2 Internal Events Results

IPPSS Plant Damage State Frequency	IPPSS Release Category/ Frequency	Revised Plant Damage State/ Frequency	Revised Release Category/ Frequency
SEFC/3.8(-5)		SEFC/7.2(-5)	
AEFC/8(-6)		AEFC/9.4(-6)	
SEC/4.1(-6)		SEC/2.2(-6)	
AEC/2.7(-8)		AEC/--	
SLFC/1.3(-5)		SLFC/1.5(-5)	
ALFC/2.1(-5)	8B/8.8(-5)	ALFC/7.8(-5)	8B/1.9(-4)
SLC/5.7(-8)		SLC/--	
ALC/4.2(-10)		ALC/--	
TEFC/3.3(-6)		TEFC/1.5(-5)	
TEC/2(-7)		TEC/--	
SEF/3.8(-9)		SEF/--	
AEF/7.6(-10)		AEF/--	
SLF/4.3(-9)	8A/1.2(-8)	SLF/--	8A/--
ALF/1.8(-9)		TEF/--	
TEF/1.4(-9)		ALF/--	
SE/1(-6)		SE/5(-7)	
SL/8.2(-11)	2RW/1.1(-6)	SL/--	2RW/6(-7)
TE/1.1(-7)		TE/--	
V/4.6(-7)	2/4.7(-7)	V/6(-7)	2/6(-7)
AE/3.2(-9)		AE/--	Z-1/--
AL/1.1(-12)	Z-1/3.2(-9)	AL/--	

Table 5.2-2. Indian Point 3 Internal Events Results

IPPSS Plant Damage State Frequency	IPPSS Release Category/ Frequency	Revised Plant Damage State/ Frequency	Revised Release Category/ Frequency
SEFC/7.9(-6) AEFC/8.7(-6) SEC/1.4(-7) AEC/3.2(-8) SLFC/8.4(-5) ALFC/2.3(-5) SLC/1.2(-9) ALC/7.7(-10) TEFC/5.2(-7) TEC/7.5(-8)	8B/1.3(-4)	SEFC/1.6(-4) AEFC/8.6(-6) SEC/-- AEC/-- SLFC/1.3(-5) SLC/-- ALC-- TEFC/7.5(-6) TEC/-- ALFC/7.8(-5)	8B/2.7(-4)
SEF/1(-9) AEF/4.2(-10) SLF/3.1(-9) ALF/8.5(-10) TEF/3.9(-10)	8A/5.8(-9)	SEF/-- AEF/-- SLF/-- ALF/-- TEF/--	8A/--
SE/6.3(-7) SL/7.9(-11) TE/7.1(-8)	2RW/7(-7)	SE/5(-7) SL/-- TE/--	2RW/6(-7)
V/4.6(-7)	2/4.8(-7)	V/7(-7)	2/7(-7)
AE/1.4(-9) AL/8.2(-10)	Z-1/2.2(-9)	AE/-- AL/--	Z-1/--

- SEFC - The Value 7.2(-5) is the summation of 5 numbers. They are:

  1) 3.8(-5) = loss of component cooling water event discussed in Section 4.6,

  2) 7.4(-6) = ATWS events discussed in Section 4.4.1,

  3) 1.5(-5) = loss of offsite power event discussed in Section 3.2.4,

  4) 2.4(-6) = loss of offsite power event discussed in Section 3.2.10,

  5) 9.3(-6) = small LOCA event discussed in Section 3.2.13.

- AEFC - The value 9.4(-6) is the summation of 3 numbers. They are:

  1) 5.6(-6) = the large LOCA event discussed in Section 3.2.11,

  2) 2.4(-6) = a medium LOCA and failure of low pressure injection (Sequence 14 on IPPSS Table 8.3-9).

  3) 1.4(-6) = a medium LOCA and failure of high pressure injection (Sequence 19 on IPPSS Table 8.3-9). High pressure injection is discussed in Section 2.4.

- SEC - The value 2.2(-6) was calculated in Section 3.2.12 and represents a loss of offsite power event.

- SLFC - The value 1.2(-5) was calculated in Section 3.2.7 and represents a small LOCA event.

- ALFC - The value 7.8(-5) is the summation of 2 numbers. They are:

  1) 3.9(-5) = the large LOCA event discussed in Section 3.2.8.

  2) 3.9(-5) = the medium LOCA event discussed in Section 3.2.9.

- TEFC - The value 1.5(-5) was calculated in Section 4.3 and represents several "feed and bleed" sequences.

- SE - The value 5(-7) was calculated in Section 3.2.14 and represent a loss of offsite power event.

- V - The value 6(-7) is the summation of 2 numbers. They are:

  1) 3.4(-7) = the interfacing systems LOCA event described in Section 3.2.15.

  2) 2.6(-7) = the steam generator tube rupture event described in Section 4.1.

<u>Unit 3 Internal Events - Table 5.2-2</u>

Via comparison, it can be noted that 7 of the 21 IPPSS damage state frequencies have been revised for Unit 3. These revisions are summarized below.

- SEFC - The value 1.6(-4) is the summation of 4 numbers. They are:

  1) 1.4(-4) = loss of component cooling water event discussed in Section 4.6.

  2) 8.6(-6) = ATWS events discussed in Section 4.4.1.

  3) 9.8(-6) = the small LOCA event discussed in Section 3.3.6.

  4) 2.4(-6) = the loss of offsite power event discussed in Section 3.3.7.

- AEFC - The value 8.6(-6) is the summation of 3 numbers. They are:

  1) 6.2(-6) = the large LOCA event discussed in Section 3.3.5,

  2) 1.8(-6) = a medium LOCA and failure of low pressure injection (sequence 9 on IPPSS Table 8.3-10). Low pressure injection is discussed in Section 4.2.

  3) 6.1(-7) = a medium LOCA and failure of high pressure injection (sequence 18 on IPPSS Table 8.3-10). High pressure injection is discussed in Section 2.4.

- SLFC - The value 1.3(-5) is the summation of 2 numbers. They are:

  1) 1.3(-5) = the small LOCA event discussed in Section 3.3.1,

  2) 6.8(-7) = a loss of main feedwater, failure of auxiliary feedwater and high pressure recirculation (sequence 14 on IPPSS Table 8.3-10). Auxiliary feedwater and high pressure recirculation are discussed in Section 2.4.

- ALFC - The value 7.8(-5) is the summation of 2 numbers. They are:

  1) 3.9(-5) = the large LOCA event discussed in Section 3.3.3.

  2) 3.9(-5) = the medium LOCA event discussed in Section 3.3.4.

- TEFC - The value 7.5(-6) was calculated in Section 4.3 and represents several "feed and bleed" sequences.

- SE - The value 5(-7) was calculated in Section 3.3.11 and represents a loss of offsite power sequence.

- V - The value 7(-7) is the summation of 2 numbers. They are:

  1) 4.6(-7) = the interfacing systems LOCA event described in Section 3.2.15.

  2) 2.4(-7) = the steam generator table rupture event described in Section 4.1.

5.2.2 External Events

Tables 5.2-3 and 5.2-4 summarize the effect that the findings discussed in the previous sections have on the Indian Point Unit 2 and 3 external event release categories. (The IPPSS did not report the external event plant damage state frequencies.)

The first column is a listing of the IPPSS significant external event release categories. The nomenclature is: Z-1Q denotes direct failure of the containment building followed by a core melt, 2RW denotes a core melt followed by a late overpressure failure of containment, 8A denotes a core melt in which the spray system fails and the containment does not fail followed by a release at the design basis leak rate, and 8B denotes a core melt in which the

Table 5.2-3

Indian Point 2 External Events Results

Release Category	IPPSS Release Category Frequency	Revised Release Category Frequency
Z-1Q	6.8(-7)	1.4(-6)
2RW	3.2(-4)	1.3(-3)
8A	4.8(-5)	4.7(-9)
8B	2.5(-6)	3.0(-5)

Table 5.2-4

Indian Point 3 External Events Results

Release Category	IPPSS Release Category Frequency	Revised Release Category Frequency
Z-1Q	3.7(-8)	7.4(-8)
2RW	6.4(-5)	2.4(-4)
8B	1.8(-6)	1.8(-5)

spray system operates and the containment does not fail followed by a release at the design basis leak rate.

Columns two and three represent the IPPSS and revised release category frequencies respectively.

Unit 2 External Events 1--Table 5.2-3

The revisions to the IPPSS Unit 2 release category frequencies are summarized below.

- Z-1Q - The value 1.4(-6) was calculated in Section 3.2.16 and represents a seismic event which causes a direct failure of containment.

- 2RW - The value 1.3(-3) is the summation of 5 numbers. They are:

  1) 4.2(-4) = the fire events leading to damage state SE discussed in Section 2.7.4,

  2) 2.3(-6) = the fire event leading to damage state TE discussed in Section 2.7.4,

  3) 2.8(-4) = the seismic event discussed in Section 3.2.1,

  4) 5.4(-4) = the hurricane event discussed in Section 3.2.5,

  5) 1.6(-5) = the tornado event discussed in Section 3.2.6.

- 8A - The value 4.7(-9) was obtained by eliminating the fire component from the external event release category 8A. This was discussed in Section 2.7.4.

- 8B - The value 3.0(-5) is the summation of 2 numbers. They are:

  1) 1.2(-5) = the fire event leading to damage state TEFC discussed in Section 2.7.4.

  2) 1.8(-5) = the fire event leading to damage state SEFC discussed in Section 2.7.4.

Unit 3 External Events - Table 5.2-4

The revisions to the IPPSS Unit 3 release category frequencies are summarized below.

- Z-1Q - The value 7.4(-8) was calculated in Section 3.3.12 and represents a seismic event which causes a direct failure of containment.

- 2RW - The value 2.4(-4) is the summation of 4 numbers. They are:

  1) 2.1(-4) = the fire event leading to damage state SE discussed in Section 2.7.4,

  2) 9.0(-6) = the fire events leading to damage state TE discussed in Section 2.7.4,

  3) 2.4(-5) = the seismic event discussed in Section 3.3.8,

  4) 9.2(-7) = the tornado event discussed in Section 3.3.9.

- 8B - The value 1.8(-5) represents the fire event leading to damage state TEFC discussed in 2.7.4.

5.2.3 Combined Internal and External Events

Tables 5.2-5 and 5.2-6 summarize the effect that the internal and external event findings have on the Unit 2 and 3 total release categories. The frequencies listed in Table 5.2-5 were obtained by summing the frequencies listed in Tables 5.2-1 and 5.2-3. The frequencies listed in Table 5.2-6 were obtained by summing the frequencies listed in Tables 5.2-2 and 5.2-4.

Via comparison of the tables, it can be noted that, at the Indian Point 2, external events comprise 100 percent of Z-1Q, >99.9 percent of 2RW, and 14 percent of 8A and 8B. Internal events comprise 100 percent of category 2, <.1 percent of category 2RW, and 86 percent of 8A and 8B.

At Indian Point 3 external events comprise 100 percent of Z-1Q, >99.9 percent of 2RW, and 6 percent of 8A and 8B. Internal events comprise 100 percent of category 2, <.1 percent of category 2RW, and 94 percent of 8A and 8B.

Table 5.2-5

Combined Indian Point 2 Internal and External Event Results

Release Category	IPPSS Release Category Frequency	Revised Release Category Frequency
Z-1Q	6.8(-7)	1.4(-6)
2	5.4(-7)	6(-7)
2RW	3.3(-4)	1.3(-3)
8A	4.8(-5)	
8B	9.1(-5)	8A+8B 2.2(-4)

Table 5.2-6

Combined Indian Point 3 Internal and External Event Results

Release Category	IPPSS Release Category Frequency	Revised Release Category Frequency
Z-1Q	3.7(-8)	7.4(-8)
2	4.9(-7)	7(-7)
2RW	6.5(-5)	2.4(-4)
8A	1.1(-6)	
8B	1.3(-4)	8A+8B 2.9(-4)

Of these five release categories, the most important in terms of risk are Z-1Q, 2 and 2RW. The revised Z-1Q and 2 categories are less than or equal to a factor of two greater than the IPPSS estimates. In the field of PRA, factors of two do not represent a significant disagreement. The revised 2RW frequency, however, is about a factor of four greater than the IPPSS estimate. This disagreement may be significant.

At Indian Point 2 the source of this disagreement is caused primarily by the SE fire sequence discussed in Section 2.7.4 and the hurricane sequence discussed in Section 3.2.5. At Indian Point 3 the source of this disagreement is caused primarily by the SE fire sequence discussed in Section 2.7.4.

In conclusion, we find the IPPSS frequency estimates for the risk significant release categories to be reasonable except for category 2RW. As mentioned above, external events comprise greater than 99.9 percent of category 2RW. We agree that external events are important to the risk of the Indian Point units; however because of the immaturity of the metodology, we do not place a great deal of confidence in the absolute value of the external event frequency estimates. We reemphasize here, as stated earlier in this report, that we believe external event frequency estimates are best used as relative values. Because of this, we feel a statement such as "at Indian Point 2, fires which lead to a 2RW radioactive material release are more probable than at Indian Point 3 by roughly a factor of 2," has more meaning than the statement "at Indian Point 2, fires lead to a 2RW release at a frequency of 4.2×10^{-4}/yr." External event analysis as applied to PRA is in its infancy. We commend the IPPSS for attacking this difficult problem, a problem which the vast majority of other PRAs did not include within their scope. However, the IPPSS external event data and the mathematical models, as well as the alternate data and models we used in this review, are somewhat simplistic.

5.2.4 Sensitivity Issues

Presented below is a summary of the results of sensitivity analyses for selected issues.

Issue	Results
1) No feed and bleed (Section 4.3)	IP 2 • Revised internal event core melt frequency increased by a factor of four. • Risk not significantly increased.

	<u>IP 3</u> • Revised internal event core melt frequency increased by less than a factor of two. • Risk not significantly increased.
2) Core melt/system interaction (Section 4.4)	<u>IP 2 and IP 3</u> • IPPSS release category frequencies not significantly affected due to influence of external events.
3) Reactor Coolant Pump Seal LOCA (Section 4.5)	<u>IP 2</u> • Revised 2RW release category and core melt frequencies reduced by a factor of 1.5. <u>IP 3</u> • Revised 2RW release category reduced by a factor of 4.5. • Revised core melt frequency reduced by a factor of 3.

LETTER REPORT
APPENDIX A

Specific comments and Questions on the Human Reliability Analysis in the Indian Point Probabilistic Safety Study

Section 2.5 of the main body of this Sandia evaluation of the IPPSS identifies and describes 11 areas related to agreement/disagreement with the human reliability analysis (HRA) in the IPPSS. This appendix lists some detailed comments not found in Section 2.5. The comments reflect the opinion of the Sandia human reliability analyst that (1) considerable conservatism occurred in several analyses in the IPPSS, (2) some judgments about human behavior, expecially behavior under stress and the number of people in the control room who would pay attention to certain detailed display indications or switch manipulations, are highly questionable, and (3) it is not possible to fully track the IPPSS HRA because of poor documentation. The comments are made according to page numbers in the document. Since the Sandia analyst also evaluated the HRA in the Zion PSS, some comparisons of the HRA in that study are made with the HRA in the IPPSS.

The comments in this appendix are restricted to an evaluation of the HRA in the probabilistic safety study in Indian Point Unit 2. As near as this analyst can judge, the HRA for Indian Point Unit 3 is identical to that in Indian Point 2.

1.0 <u>Comments on Volume 1</u>

1.1 Section 0.15 "Human Error Rates" is essentially identical with the equivalent section in the Zion study (ZPSS). The IPPSS also states that the 90th percentile will be used as the upper error bound, whereas the 95th was used. For the most part, the IPPSS human error rates (HERs) are calculated from human error probabilities (HEPs) from the Handbook*, with the correct assumption that these HEPs are medians of log normal distributions.

1.2 Section 0.16.9 "Human Error Contributions to System Failure Rate." The equations are correct for levels of dependence, but in some cases this analyst believes that the IPPSS assesses inappropriate dependence levels, as discussed later.

1.3 Page 1.3-120 "K-4 Manually Deenergize and Rods Drop." This assessment is evaluated in Section 2.5. The IPPSS HRA appears to be overly optimistic.

*Swain, A. D. and Guttmann, H. E., <u>Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications</u>, NUREG/CR-1278, U.S. Nuclear Regulatory Commission, Washington, DC, October 1980.

1.4 Page 1.3-127, Section 1.3.3.9 "Operator Actions (OP)."

1.4.1 OP-1 Primary Bleed is used in several event trees (which are listed on page 1.3-152). OP-1 is used specifically in ET-3 Small LOCA, ET-5 Steam Break Inside Containment, and ET-6 Steam Break Outside Containment. The IPPSS uses their "High Pressure Recirculation Model" for human performance assessments. Their mean HER is 3.46×10^{-4} as compared with the ZPSS mean HER of 1.3×10^{-4} for this model. The difference is the assignment of moderate dependence of the shift technical advisor (STA) with the rest of the control room personnel as compared with the assessment of low dependence for the STA in the Zion plant. This difference is correctly justified and explained. However, a major bit of optimism may occur in the IPPSS assumption that the STA will bother with details of feed and bleed as well with the details of other monitoring and switching actions.

1.4.2 OP-2 Operator Feed and Bleed is used in ET-7 Loss of Main Feedwater, ET-9 Loss of Primary Flow, ET-10 Core Power Excursion, ET-11 Turbine Trip, and ET-12 Reactor Trip. It appears that OP-2 reduces to OP-1 for the calculations in the IPPSS.

1.4.3 OP-3 Operator Controls Transient is used in ET-8 Closure of One Main Steam Isolation Valve (MSIV) and ET-10 Core Power Excursion. For ET-8, OP-3 is assigned an unavailability of 0. This assessment may be optimistic--what are the cues to the operator that ET-8 has occurred, as there is no reactor trip? On page 1.3-207 it is stated, "An operator should respond to increasing T_{avg} by searching for the cause of the transient. Until the cause is determined, the effects can be opposed by negative reactivity insertion. Once identified, the cause can be terminated." Assuming that it is an analog meter that displays T_{avg}, what cues direct the operator's attention to the meter? Unless it is a rather potent cue, e.g., an item to be checked off on the shiftly audit of safety-related equipment, the probability of detection could be very small.

For ET-10, the IPPSS says that less than 15 minutes is required for operator action, yet the required operator performance is assessed using the HP Recirculation Model, which is 2 hours into the transient on which this model is based. This appears to be a gross misapplication of the model.

1.4.4 OP-4 Operator Depressurize and Stablize is used for ET-4 Steam Generator Tube Break. Action is required in the first 30 to 45 minutes, and the IPPSS uses OP-4 = OP-2, i.e., the HP Recirculation Model. Note that the time requirements for this application differ from the time on which the model is based.

1.4.5 OP-5 Primary Cooling Feed and Bleed with Emergency Boration is used in ET-13 Anticipated Transient Without Scram (ATWS). Apparently there are several options for the operator, including a possible requirement to manually start the safety injection pumps and perform the correct alignment of valves within 25 minutes. The IPPSS assumes a high level of stress so the Low Pressure Recirculation Model is used. As in the ZPSS, if ATWS is combined with other failures, the use of this model will give a joint HER for the control room personnel wihch is far too optimistic. Also, use of this model for ET-13 presumes that the STA would be involved in the details related to OP-5.

1.5 On page 1.3-149, ET-13 Failure of the Reactor to Trip (i.e., ATWS) is discussed. This ET is placed in several of the other event trees, and this means that the use of OP-5 for the human error contribution is very optimistic since OP-5 is based on ATWS alone. It appears that all of the 5 OP terms ignore the effects of multiple faults on the stress level of the operating team, and also that the STA is always involved in switching details.

1.6 Page 1.3-219, "ET-11b Turbine Trip/Loss of Offsite Power." How can OP-2 be employed without AC power? The event tree is not clear to this analyst.

2.0 <u>Comments on Volume 2</u>

2.1 In Section 1.3 the five OP disgnators are used in many tables. This analyst cannot understand the variability of the mean HERs for the various OP designators when they are used in different event trees. For example, on page 1.3-628, why is OP-5 HER = 8.9×10^{-3} for ET-13 instead of the basic HER of 4.75×10^{-3} from Volume 1? This is an illustration of the difficulty in tracking the HER estimates given in the IPPSS.

2.2 Page 1.5-121, Section 1.5.1.4.1 "Basic Human Error Rates." This section is essentially (if not actually) a repeat of the equivalent section in the ZPSS.

2.3 Page 1.5-123 provides the within-person dependence assumed for valve restoration evaluated in Section 2.5 of the main body of the Sandia report. The possibility of optimism is stated in that section.

2.4 Page 1.5-123. The IPPSS (like the ZPSS) assumes that all procedures used in Indian Point consitute a short list (i.e., 10 or fewer items) and that checkoff provision is always required and always used properly. These assumptions will nearly always result in optimistic evaluations of tasks requiring the use of procedures. However, in all but two cases in which the HRA is based on the use of written procedures, the IPPSS actually used a more conservative .003 as the basic HER. This is a factor of 3 higher than the basic HER presuming a short list and checkoff provi- sion always used properly. .003 is the baisc HEP for NUREG/CR-1278 (Table 20-20) for errors of omission using either (1) a long list with checkoff provisions always used properly or (2) a short list with no checkoff provi- sions (or with checkoff provisions used improperly).

2.5 Page 1.5-125, Section 1.5.1.4.3 "High Stress Situations." It is stated that the IPPSS considers the time available and the information the operators have when an analysis is made for high stress situations. But their implementation of this principle is invalid when the LP or HP recirculation models are applied without modification to cases with multiple problems or with severe time constraints.

2.6 Calculations related to diesels.

2.6.1 Page 1.5-170. In describing unavailability of the diesel generators, the IPPSS notes that an alarm is received in the control room if the AUTO/OFF/MANUAL deisel generator switch (in the diesel generator room) is removed from the AUTO position. If this plant operates like other plants, the control room would be notified prior to someone turning off the switch to permit maintenance, etc. The alarm would come on, and the control board operator would glance up to verify it is the diesel alarm, and then cancel the alarm, which would then remain as a steady-on white annunicator tile. When the maintenance, or whatever, is finished, the technician would return the AUTO/OFF/MANUAL switch to AUTO, and the steady-on indication of the tile in the control room would go off. (The IPPSS does not indicate if the tile would just turn off, or if there would be some kind of flashing with or without an auditory signal.)

The above type of alarm can provide an excellant recovery cue if the plant uses a good administrative procedure. However, if the technician forgets to throw the switch to the correct position, the only cue to the operators is the steady-on condition of the annunciator tile. The IPPSS incorrectly states (page 1.5-188) that the Handbook does not address the issue of an operator failing to respond to an annunciator which stays lighted for some time following its expected receipt." Reference to Table 20-12 in the Handbook indicates that the HEP is very high if no special checklist is used which specifically has the annunicator tile in question as an item to be checked off. For noticing one of the usual several steady-on annunciator tiles, which are no longer annunciating, the estimated HEP is .9 for the initial scan in a shift and .95 for the 7 presumed remaining scans during the shift. If only these values were to be used, and if one assumes that the first opportunity for the control board operator to make this detection occurs midway into the shift, the probability of failing to detect the annunciator wuold be $.94^4 = .81$ for the remainder of the shift and for 12 hours it would be $.95^4 \times .9 \times .95^7 \approx .5$. This type of analysis would be valid only if the Indian Point plant has a very poor administrative control procedure. A good control procedure which involves the use of a written checklist after the diesel has been (presumably) placed back on line could reduce the above HEP by a sizeable amount, and the Handbook covers this possibility, too.

2.6.2 On Page 1.5-188, the IPPSS develops (on some unstated basis) a frequency estimate of recovery probabilities versus time. For example, they assume that within 1 hour, the error of omission (of forgetting to place the switch back in the AUTO position) would be recovered with probability of .75. One presumes that there must be some kind of good administrative control procedure at the plant to warrant such a recovery probability, but the IPPSS does not state what it is. Also the IPPSS does not describe how the estimates of time to restoration of the switch were derived. This analyst presumes that their numbers are reasonably conservative, but it would have been better if they had performed the appropriate analytical study and developed and described the recovery estimates based on the recovery factors at the plant.

2.6.3 On page 1.5-189, the IPPSS states that operators who begin a new shift "are likely to notice the annunciator during their review of the control boards." This analyst does not concur with this statement for reasons expressed above, unless one defines a .5 probability of recovery in 12 hours as "likely."

2.6.4 At the bottom of the same page, the IPPSS provides a mean estimate for the unavailability of a diesel generator due to its control switch being left in the wrong position at the completion of a monthly test. This analyst cannot figure out how the mean was obtained.

2.6.5 In paragraph 2, page 1.5-190, the IPPSS assigns moderate dependence between the errors of leaving the 3 diesels in other than the AUTO position. Their rationale is spelled out, and seems conservative. However, to properly evaluate this dependence assessment, it would be necessary to look at the written procedures actually used, and understand the sequence of operation fully. For example, if the technician performing the monthly test on the 3 diesels characteristically restores all three switches to AUTO at the same time, the dependence might be complete rather than moderate.

In the same paragraph regarding the steady-on annunciator tiles signaling the non-AUTO positions of the 3 diesels during test, it is stated that "the control room operator may decide to have them all cleared at the end of the test, he may be distracted by other operations, etc. Therefore, the overall effect of the combined actions of the local test personnel and the control room operators is a low dependence between the unavailability of one diesel generator due to a mispositioned control switch and the unavailability of two or three diesel generators." This analyst cannot understand this assessment. Also, is it possible for the control board operator to clear (turn off) the steady-on indication of these annunciator tiles when the diesels are still in an off-normal condition? This analyst also cannot evaluate the unavailability estimates made on this page and on the next page (1.5-191). The final estimate for the unavailability of all three diesels due to the mispositioned switches is extremely low--10^{-7}. This analyst regards such low numbers with skepticism.

2.6.6 Page 1.5-192 evaluates human errors related to diesel fuel oil supply failure. On the following page a mean of 2.2×10^{-3} is assessed for omission of procedural step. From page 1.5-126, this assumes a short list and a checkoff always used correctly, and is based on the Handbook estimate of an HEP of .001. On the other hand, if a long list were used without a checkoff, the HEP would be a factor of 10 higher, with about a corresponding increase of a factor of 10 in the IPPSS mean HER.

2.6.7 At the bottom of page 1.5-193 there is some inappropriate reasoning about the psychology of human behavior. The IPPSS reasonably assumes that for this event the operators are performing under a moderate level of stress but the report goes on to say, "The high priorities assigned to restoration of power will tend to counterbalance these stress effects for this specific action." In other words, the report assumes that if something is important enough, then the required action will not suffer from operator stress. Studies of behavior under stress (as described in Chapter 17 of the Handbook) do not confirm this optimism.

However, at the top of page 1.5-194 the report notes that the operator has from 45 minutes to an hour to accomplish the task. Therefore, they use the mean HER unmodified for stress. Even though this analyst regards part of their logic as faulty, the fact that the time requirements are so generous would lead him to agree that the mean HER unmodified for stress is probably OK. The only question, then, is the assessment of an HER of 2.2×10^{-3} based on the use of the short list and 100 percent correct use of a checkoff provision.

In discussing a recovery factor for this error, the second paragraph on page 1.5-194 presents a reasonable approach and derives an estimated HER of about .008. Therefore, the equation for P_{NR} is reasonable if one accepts the value of P_{PR} based on a short list with 100 percent checkoff. (Note: the equation has a typo; the second term should be P_{PR}.) However, this analyst cannot evaluate whether or not 20 to 25 minutes is sufficient time for someone (after being cued by the annunicator) to restore power to the Motor Control Center (MCC) before the diesel task is completely drained. It sounds like that should be enough time, if one can assume that it is completely obvious what to do when the annunciator sounds.

2.6.8 On page 1.5-219, Section 1.5.2.2.1.4.6.2. "Human Error," reference is made to the above calculations. Then a new HER of 9×10^{-3} is quoted for "...errors made by maintenance personnel during the disassembly, inspection, repair, and reassembly of the diesel engine, generator, or any portion of their control or support systems." It is stated that this HER comes from item 1 in the table on page 1.5-126, but this Handbook value is not appropriate for the human actions quoted. In fact, the Handbook simply does not cover this aspect of maintenance because it is assumed that such errors are included in the usual equipment reliability figures, and one should not count failures twice. Nevertheless, the HER is a very conservative number, almost 10^{-2}, and a very, very conservative assumption is made in the IPPSS that any such error would result in a failure of the diesel. Since diesels are tested after maintenance, they assume that the test would not catch 5 percent of these errors. So on page 1.5-220, the report assumes an HER of 4.5×10^{-4} undetected error per maintenance event. This seems very conservative to this analyst.

The IPPSS goes on to assume low dependence between maintenance error on different diesels. Again, this seems conservative; this analyst would probably assume zero dependence unless something in the written procedures or schedule indicates otherwise.

2.6.9 On page 1.5-222, second paragraph, it is assumed that the operator recovery rate for portions of the electrical system which are deenergized by "other independent failures following event initiation" is unity over the 6-hour period following event initiation. Does this mean that beginning with time zero at the event, the operator success rate is a linear line to a value of 1.0 at 6 hours? In other paragraphs, it is stated that no credit is given for operator recovery. This whole section is not clear to this analyst.

2.7 The general impression one gets is that the IPPSS PRA team deliberately tried to be conservative in their treatment of human errors. In some cases some incorrect logic and assumptions about human behavior were used, but there is so much conservatism, one gets the impression that the overall human reliability analysis is rather conservative.

3.0 Comments on Volume 3

3.1 On page 1.5-343, a table presents estimated action times (not including recognition and evaluation time) for several recovery actions for loss of offsite power at Indian Point Unit 2. This analyst was not able to find how these values were derived.

3.2 Reactor Protection System (RPS)

3.2.1 On page 1.5-373 in discussing the RPS, the report notes that the analysis is carried out assuming that no operator action is taken to scram the plant, yet near the bottom of the page it says, "Operator action to manually scram the plant is successful because the manual scram switch bypasses all logic channel failures." This is utterly confusing to this analyst. On page 1.5-378, Section 1.5.2.2.2.2.7, the report again says that operator scram is not included in the analysis.

3.2.2 On page 1.5-389, the report states, "If the value of a single instrument failing, 2.66×10^{-4}, is taken as the frequency of common cause miscalibration of a set of instruments, failure of two sets of instruments due to miscalibration of this type would result in a mean" of 4.61×10^{-7}. This statement seems to imply that an equipment failure rate is used as the estimate of a human error common cause factor. Is this correct? If so, what is the rationale?

3.2.3 In the same paragraph it is stated that "most calibration activities, even if performed in error, do not result in an instrument that fails to provide a trip." This analyst cannot understand the implication of this statement. If the instrumentation technician has incorrectly set up the equipment he is using to calibrate some set points, it is quite possible that his error would serve as a common cause condition that might result in much too high a trip set point for several redundant channels, with a reduction in safety protection.

3.3 Safeguards Actuation System (SAS) (Safety Injection and Containment Spray Actuation)

3.3.1 On page 1.5-415, Section 1.5.2.2.3.2.4 "Operator Interaction," the report states that operator action to manually initiate the SAS is excluded from the analysis. If this means that no operator credit is allowed for manual initiation, this is a reasonable bit of conservatism due to the reported reluctance of some operators to take a chance and erroneously activate the SAS, with the possibility of severe reprimands or worse.

3.3.2 On page 1.5-419, it is apparently assumed that if there is a failed DC power fuse, it will be detected 100 percent of the time during the operator check of the control room panels at the beginning of the shift. On what basis is a zero probability of error assessed? What are the cues to the operator? Is there a written checklist and does this particular fuse have to be checked off? Even if the answer is yes, a probability of zero is not believable.

3.3.3 On page 1.5-421, the same assumption of operator infallibility is made for the pressurizer pressure transmitters.

3.3.4 On page 1.5-422, it is estimated ("based on engineering judgment") that the mean detection time for failure of the Containment Pressure Transmitters is 24 hours. On what basis? What are the cues for operator detection?

3.3.5 Page 1.5-422, Section 1.5.2.2.3.4.5 Quantification of Common Cause notes that there is a potential for human error resulting in miscalibration of several set points. This possibility is given a very low probability of 2.94×10^{-10}. If this is a human error failure, it is implausible. Also, the use of equipment failure rates to estimate common cause due to human error is not given a justification.

3.3.6 Same page, Section 1.5.2.2.3.4.6. Quantification of Human Error. Two human errors are given a zero probability: (1) failure to return "Normal-Defeat" switch to "Normal" after testing, and (2) failure to return individual test switches to "Normal." The rational given is that these switches are annunciated when not in the "Normal" positions. But the usual plant procedure would be that the operators in the control room would turn off the sound and the flashing of the annunciator tiles, and then the steady-on tiles would merge into the usual background of at least 20 tiles that are steady-on. If this is the only recovery cue, it is not a good one. Perhaps there are some administrative recovery procedures as described in Section 2.5 of the main body of the Sandia report. Otherwise, the IPPSS assessments may be optimistic.

3.4 High Pressure Injection System (HPIS)

3.4.1 On page 1.5-480, Section 1.5.2.3.1.4.1.3 Human Error Contribution. The rationale for assessing no significant human error due to flow tests, etc., seems reasonable to this analyst.

3.5 Low Pressure Injection System (LPIS)

3.5.1 The same argument (page 1.5-522) is used for LPIS unavailability due to human error, and seems reasonable. However, in both cases one might ask if there are any blocking valves that are locally operated and which do not receive signals to open upon safety injection.

3.6 Accumulators

3.6.1 On page 1.5-556, no human error contribution to the unavailability of the accumulator is assessed because it is passive, and the accumulator level is annunicated if it is too high or too low. In the latter case, the operators should respond quickly. The assessment seems reasonable.

3.7 High Pressure Recirculation (HPR) (Small LOCA)

3.7.1 The two major human errors having impact in the systems analysis (failure to initiate HPR and erroneous operation of switch 7) are evaluated in Section 2.5 of the main body of the Sandia report.

3.7.2 In paragraph 3, page 1.5-587, it is noted that if there is no flow from the recirculation pumps, the operators can switch to RHR pumps to establish recirculation flow. They correctly assess a high level of stress for these activities, and use a basic .1 HEP from the stress curve in the Handbook. The error is attributed to all 4 people, including the STA. This seems reasonable since a no-flow situation would demand the attention of everyone. Since recirculation is not required until at least 2 hours into the accident and since the time window for switchover is 60 minutes, the .1 HEP may seem unduely conservative. However, countering this is the fact that the procedure will be required only if something everyone expects to work properly does not do so. Under these conditions, one would expect the stress level to rise rapidly. Therefore, in the opinion of this analyst the assessment of .1 seems reasonable as the basic HEP for one person even though the time allowances are not stressful. Using the usual assessments of high dependence for the second operator, and moderate dependence for both the watch supervisor and shift technical advisor, the IPPSS calculates the joint median HEP is 2.9×10^{-3}, with a calculated mean of 1.52×10^{-2} assuming a lognormal distribution and an error factor of 20. The wide error bounds seems reasonable under this case of stress.

3.7.3 Same page. The above calculation is also assigned to the case of failures of the train consisting of heat exchanger 21 (but not the HX itself) and the adjacent valves. In this case the operators should open MOVs 746 and 747 to establish flow from HX22 to the suction side of the SI pumps. The assessment seems reasonable to this analyst.

3.8 Low Pressure Recirculation (LPR) (Large LOCA)

3.8.1 The two major human errors having impact in the systems analysis (failure to initiate LPR and erroneous operation of switch 6) are evaluated in Section 2.5 of the main body of this report

3.8.2 At the bottom of page 1.5-603 some reasonable assumptions are made about the extremely high stress level that would occur if the recirculation pumps were not available (or failed quickly after recirculation) and the operators then had to switch to the RHR pumps and establish or maintain recirculation flow from the containment sump. A basic HEP of .5 is assessed. They reasonably assume that all four persons in the control room would be involved in this unusual event, and assign high dependence among all of them. The .5 basic HEP for one person then translates to a .21 joint median HEP which is about the same value one would obtain from this analyst's latest model.* An HER of .26 is calculated using an EF of three. This analyst would use an EF of five, with the upper bound of 1.0. This would increase the HER somewhat.

3.8.3 On page 1.5-607, the loss of electric bus 6A is judged to be an extremely high-stress situation (from the Handbook, page 17-19), and the basic HEP for an operator to restore the bus is given as .25. However, the document assumes that the bus would fail at safeguards actuation. Reference to the Large LOCA curve in the Handbook indicates that the basic HEP would be very high right after the Large LOCA occurs (when safeguards actuation would occur), and would not decrease to the .25 level until about 25 minutes into the LOCA. But the report assumes that it would be possible to restore bus 6A either from the control room or locally before switchover is needed (at 20 to 40 minutes into the LOCA). However, as near as this analyst can determine, no credit was given in the IPPSS for restoration of bus 6A, so perhaps the question is moot. If this is not the case, some recalculations are in order.

3.8.4 At the bottom of this page, a basic HEP of .25 is judged to be appropriate for actions under the above partial panel operation. In the case of failure to initiate switchover, this .25 is halved, based on the assumed high level of familiarity of the control room personnel with this Large LOCA requirement. This seems like a reasonable argument, and is evidence of the effort by the IPPSS analysts to avoid optimism.

3.8.5 On page 1.5-608, the error on switch 6 (discussed above in paragraph 3.8.2) is reassessed under the above partial panel operation. However, in this case the .25 basic HEP is not divided by two, a reasonable assumption, and moderate rather than low dependence is assigned to the STA for the recovery factor. This analyst still questions the use of all four people for the correction factor for the error on switch 6. If the same assumptions are employed as were used for LPR, the joint HEP will remain .156, but the recovery factor failure becomes: $\left[\frac{1 + 6 \times .25}{7}\right]^2 = .128$, and the

joint unrecovered HEP becomes $.128 \times .156 = .02$, rather than the 3.11×10^{-3} calculated in the IPPSS. If this factor of 10 increase would apply (roughly) to their calculated HER, it would then be about .1 rather than about .01.

3.9 Containment Spray Recirculation (CSR)

3.9.1 On page 1.5-614, the IPPSS assesses an HER of 1.5×10^{-3} for establishing CSR. The cues are obvious--a rise in containment pressure after core cooling recirculation has been established using the regular procedures for a LOCA. However, the Sandia systems analysts have determined that there is no written procedure for refilling the RWST (which would be required to get water to the containment spray system). Therefore, the IPPSS use of a basic .1 as the HEP may be optimistic. However, it appears that no credit is given for CSR anyway. This is discussed further in Section 2.2 in the main body of the Sandia report.

3.9.2 On page 1.5-616 the same CSR is discussed following loss of electric bus 5A or 6A. The IPPSS makes the judgment that the operators' stress levels would not be increased by the need to use CSR after the unexpected failures of bus 5A or 6A. Their argument is the fact that the operators would have already established core cooling recirculting using LOCA procedures would have "some reassuring effect on the operators." This seems like a questionable argument to this analyst, but more information would be needed, and the whole issue of credit for CSR is discussed in Section 2.2 in the main body of the Sandia report.

3.10 Containment Spray Injection (CSI)

3.10.1 On page 1.5-702, Section 1.5.2.3.5.2.4.4, Manual Operator Action, the report states, "If the operator determines that the accident is a steam break, he has 2 minutes to press the 'NaOH' button on the safeguards panel." Giving any operator credit for this action is highly questionable, and this analyst presumes no such credit was given in the IPPSS.

3.10.2 On page 1.5-711, operator errors are considered for failing to reopen manual containment isolation valve 869A or 869B for the pump train under test.

The IPPSS uses an inappropriate HER of 9×10^{-3} which is based on a .005 basic HEP from their table on page 1.5-126 (taken from the Handbook). Actually, a basic HEP of .003 (or probably less) would be more appropriate, because it is presumed that the restoration procedures are carried out with some kind of written procedure. (The .003 value is based on a long list with checkoff properly used or a short list with no checkoff used--from Table 20-20 in the Handbook.) Therefore, the basic HEP used in the IPPSS is probably a factor of two conservative.

The report then assumes a recovery factor (with an HER of 2.2×10^{-2}, based on a Handbook basic HEP of .01 from Table 20-21) which implies there is a second person who goes to the valves to check that the original operator restored (or did not restore) the valves. This is not the usual plant practice, and this analyst assumes that the IPPSS analysts did determine that this excellent recovery practice is indeed followed at the plant.

The IPPSS assumes a low level of dependence between errors of omission for the two valves, but does not give a basis for this assumption. If the valves are next to each other, for example, complete dependence might be the appropriate assessment. The document assumes independence for errors of omission during the recovery inspection; again no rationale is given. The level of dependence might well be some nonzero level of dependence, especially since the original error assumes a nonzero level of dependence. In fact, given that two valves were not restored and given that the checker discovers one, he is almost certain to discover the other nonrestored valve.

The $Q_{Human\ Error}$ equation reduces to $P_{(both}P_{(nondiscovery)}$ because the other error term is multiplied by $P_{(pump\ train)}$ which apparently is 6.9×10^{-3} for one train (from the previous page). Therefore, $Q_H = .009 \times .05 \times .022$ which is 9.9×10^{-6}. This analyst does not understand how they got 1.4×10^{-5}. It probably doesn't matter since 1.4×10^{-5} is such a small number.

3.10.3 On page 1.5-715 a mean NA_H of 2×10^{-4} is given as the human error contribution to the system unavailability. This analyst cannot determine how this value was derived.

3.11 Containment Fan Cooling System

3.11.1 On pages 1.5-749 and 755 it is noted that, "Failure of the automatic start system can be compensated for by operator action. There are sufficient indicators and alarms so that the operator would be immediately aware that the units had not started properly or the valves had not switched to their accident position, and he would start the units or open the valves manually." This statement apparently ignores the very high level of stress right after a Large LOCA when the fan coolers should start automatically. However, the IPPSS does not include the effects of human interaction as a recovery factor, so the optimistic statement above is immaterial.

3.12 Component Cooling System

3.12.1 On page 1.5-769 it is stated that no human error contribution for recovery is considered.

3.13 Service Water System (SWS)

3.13.1 On page 1.5-819, the IPPSS includes some calculations on "operator interaction with the SWS." This analyst cannot understand the logic used. For one thing, the <u>April</u> issue of the Handbook is used rather than the current October issue.* Also some typos make it impossible to determine what HEP estimates were used from the April issue of the Handbook. For example there is no item 21 in Table 20-7 of that issue of the Handbook. The major point, however, is that the exact nature of the error and its recovery is not stated clearly enough for this analyst to suggest a different assessment. It appears that the report may be giving credit for noticing one annunciator among many that come on after a LOCA. Or perhaps the annunciator in question is one that occurs shortly after the error of mispositioning the mode selector switch. In this case, the stress would be at an optimum level. The description of the error and its recovery factors are not clear.

3.14 Auxiliary Feedwater System (AFWS)

*Anyone with the April 1980 issue of the Handbook should discard it.

3.14.1 On page 1.5-901 it is noted that each AFWS pump is tested monthly. For this the manual gate valves in the pump discharge lines are closed. After the test they must be reopened. Is this possibility for human error considered in the IPPSS? (Presumably these valves are locally operated.)

3.14.2 On page 1.5-904 two recovery factors by the operator are listed, (1) Manually start the turbine-driven AFSW pumps, and (2) manually jack open valves PCV-1187, 1188, and 1189 to switch over to city water. Because the latter valves do not have hand cranks, the IPPSS conservatively assumes that there would not be enough time to open then in case no water from the condensate storage tank is available for AFWS (due to failure of either CST outlet valve). However, for the first recovery factor, frequency distribution histograms are presented to show mean failure probability for actions inside the control room and actions outside the control room plotted against percentage. It is stated in the document that "The probability of human inaction has been quantified into histographs based on discussions with operators, supervisory personnel, engineers, and after a review of the operating histories at other plants. The judgments take into account the high-stress conditions in the control room during emergencies and the competing demands during the 30 minutes the operator has to perform the task."

There are several problems with this approach. First, the judgments required are very complex, and should be very unreliable. The variation of judges is very likely to be extremely large. Expert opinion is best used in pair-comparing types of judgments.* Second, no documentation is provided for the techniques used for the type of psychological scaling used. This analyst can have no confidence in data derived in this manner. Third, the histograms are identical to those used in the ZPSS, so it is very unlikely that the IPPSS analysts repeated the methodology at Indian Point and happened to come up with identical results.

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*Stillwell, W. G., Seaver, D. A., and Schwartz, J. P., Expert Estimation of Human Error Probabilities in NUclear Power Plant Operations: A Review of Probability Assessment and Scaling, NUREG/CR-2255, U. S. Nuclear Regulatory Commission, Washington, D. C., May 1982.

If the data in these histograms are used in any of the important systems analysis, the estimated HERs should be doubled and quadrupled to see if this would make any difference in the results.

4.0 <u>Comments Indian Point Unit Three Probabilistic Safety Study Six</u>

4.1 This analyst made several checks on the estimated HERs used for various tasks for IP Unit Three and discovered that they are identical to those given in the analysis for IP Unit Two. Therefore, it is presumed there are no differences and that the IPPSS analysts judged that the behaviorial similarities were sufficiently high that no material amount of error in the safety study would occur if the analysis from IP2 was applied without adjustment to the same tasks in IP3. This analyst does not criticize this approach; in view of the subjectivity of much human performance data, such generalization can be warranted.

LETTER REPORT
APPENDIX B

DRAFT REPORT

REVIEW OF THE INDIAN POINT PROBABILISTIC SAFETY STUDY SEISMIC, FLOODING, AND WIND

Prepared for
Sandia National Laboratories
Albuquerque, New Mexico
July 19, 1982

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REVIEW OF INDIAN POINT PROBABILISTIC SAFETY STUDY

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1. INTRODUCTION

Jack R. Benjamin and Associates, Inc. (JBA) was retained by Sandia National Laboratories (Sandia), Albuquerque, New Mexico, to perform an in-depth review of the following sections of the Indian Point Probabilistic Safety Study (referred to as the IPPSS report), prepared for Consolidated Edison Co. of New York, Inc. (owner of Unit 2), and the Power Authority of the State of New York (owner of Unit 3), Copyright 1982.

- 7.2 Seismic
- 7.4 Flooding
- 7.5 Winds and Wind Induced Missiles
- 7.9.1 Dames and Moore Seismicity Study
- 7.9.2 Woodward-Clyde Seismicity Study
- 7.9.3 Structural Mechanics Associates, Inc., Fragility Study
- 7.9.4 Structural Mechanics Associates, Inc., Damage Effective Ground Acceleration
- 7.9.5 Research Triangle Institute Extreme Wind Analysis
- 8.3.4 Identification of Major Scenarios, Systems, and Structures Contributing to Risk - Indian Point 2 (Seismic and Wind)
- 8.3.5 Identification of Major Scenarios, Systems, and Structures Contributing to Risk - Indian Point 3 (Seismic and Wind)

These sections present the results of the analysis for Units 2 and 3 for seismic, flooding, and wind external events, and flooding internal events. Both the development of hazard and fragility curves as well as the integration leading to unconditioned core melt and release category frequencies were reviewed.

As an aid in the review, a seismic hazard model was developed to investigate the assumptions of varying the parameters leading to the frequency of occurrence of ground motion. Considerable interest concerning the effects of a postulated Ramapo fault zone has been expressed by the various reviewers. As an aid in determining the impact of a Ramapo fault zone, seismic hazard analyses were conducted for a range of assumptions.

As part of the review, a meeting was held with Pickard, Lowe, and Garrick (PLG), who prepared the IPPSS report, to discuss questions which arose from the review. At the first session the seismic and flood analyses were discussed. Woodward-Clyde Consultants (WCC) and the engineer who performed the analysis for Dames and Moore (D&M) represented PLG in the area of seismic hazard curve development. Structural Mechanics Associates (SMA), who was the seismic fragility consultant, also participated in the meeting. Pickard, Lowe, and Garrick performed the flood analysis themselves.

At the second session of the PLG meeting, the analysis for the effects of wind was discussed. Research Triangle Institute, who developed the hazard curves for hurricane and extratropical winds, and tornado, were represented. The fragility curves for wind loading were developed directly by PLG.

In addition to attending the PLG meeting, Dr. John W. Reed and Dr. Martin W. McCann of JBA visited the Indian Point site and spent one day each touring the Unit 2 and 3 plants. The purpose of the visit was to acquaint the reviewers with the plants and the safety-related equipment and structures. Dr. Reed directed the review of the IPPSS. Dr. McCann assisted in the review concentrating primarily on the seismic hazard analysis and the analysis for flooding.

Three consultants to JBA provided additional review of the IPPSS report. Professor Ronald L. Street reviewed the development of the seismic hazard curves from the seismologist's viewpoint. Professor Erik H. Vanmarcke also reviewed the seismic hazard curves. Dr. Larry R. Russell performed the review of the hurricane hazard curves. Reports from the three consultants are included as appendices to this report.

The remaining chapters in this report discuss the review of the overall methodology, provide review of specific IPPSS report sections, discuss our seismic hazard analysis, and end with the final conclusions of the review and recommendations. These chapters are entitled:

2. Overall Methodology
3. Report Sections

4. Seismic Hazard Analysis
5. Conclusions and Recommendations

The remaining sections of this chapter describe the approach used to review the IPPSS report and present the results of a sensitivity study which was conducted to gain insight into the seismic hazard and fragility curves.

In order to avoid confusion in reading this report, the chapter sections are not numbered. The figures, tables, and references are each numbered consecutively in each chapter. In contrast, sections, figures, tables, references, and pages of the IPPSS report have a decimal (or sometimes dashed) numbering system. By organizing the review report in this manner, references to the locations of material in the IPPSS report and in this report are more obvious.

REVIEW APPROACH

A dual approach was used to review the IPPSS report. One part consisted of systematically reading, reviewing, and commenting on the sections of the IPPSS report. In the second part, the review consisted of a continuous search for the parameters, assumptions, etc., which control or contributed significantly to the results of the analysis. As part of this effort, a sensitivity study for the seismic effects was conducted to determine how the mean frequency of release category 2RW changes as the relationship between the hazard and fragility curves is varied. Using this procedure, structures and equipment which contributed significantly to the frequency of 2RW were identified. Our review concentrated more heavily on the major contributors. Comments concerning the integration of the wind hazard and fragility curves are made for IPPSS Section 7.5.

The Seismic Safety Margins Research Program (SSMRP) being conducted by the Lawrence Livermore National Laboratory (LLNL) for the Nuclear Regulatory Commission (USNRC) is currently developing a procedure for estimating the risk of an earthquake-caused radioactive release from commercial nuclear power plants. Zion Nuclear Generation Station has been used as a model facility for the development of the SSMRP methodology. We have utilized the results which have been published to date for the SSMRP in our review of the IPPSS report. It should be noted that the engineers who contributed to the development of fragility data for the SSMRP are the same professionals who performed the fragility analyses for the IPPSS report. In this sense, the results of the SSMRP are not an independent comparison of the IPPSS results. However, numerous detailed analyses of the structures and probabilistic sensitivity studies have been performed in the SSMRP, which provide an independent indication of the appropriateness of some of the assumptions made in the IPPSS study.

In our review, we have attempted to make comments on both minor and major issues, looking for both conservative and unconservative assumptions. In order to help the reader and to maintain perspective ourselves, we have tried to indicate, where possible, the ultimate impact of the issues which we have raised. As an aid in doing this we have selected the mean frequency of core melt or the important release categories as the basis for comparison. We have adopted the following scale to quantify our comments in reviewing the IPPSS report:

Comment	Effect on Mean Frequency of Consequences or Core Melt
Small	Factor ≤ 2
Moderate	$2 < \text{Factor} \leq 10$
Large	Factor > 10

We have indicated in our report in several places where effects of changes in parameters will have a greater effect on the tails of the frequency of core melt or release category density functions. In general, we expect a greater impact on the tails as compared to the mean frequency; however, we feel that the mean frequency is a more important parameter in the IPPSS study.

SENSITIVITY ANALYSIS FOR SEISMIC EFFECTS

In order to understand how changes in the analysis parameters might affect the mean frequency of release category 2RW, we integrated the hazard and fragility curves using the same discrete probability distribution procedure used in the IPPSS report. The mean frequency values given in the report for 2RW are 1.4×10^{-4} per year for Unit 2 and 2.4×10^{-6} per year for Unit 3, which were used for comparison.

The hazard curves from IPPSS report Sections 7.9.1 and 7.9.2 were used in the sensitivity analysis. The relative weights which were assigned were the same as used by PLG (see discussion for IPPSS report Section 7.2). The fragility curve values for release category 2RW were obtained from Table 7.2-4 for Unit 2 and Table 7.2-8 for Unit 3.

The purpose of the sensitivity study was to determine the differences between the D&M and the WCC seismicity curves and to investigate the effects of shifting and truncating the curves. The D&M curves were shifted by a factor of 1.23 (this was done to convert from peak ground acceleration to damage-effective ground acceleration) and truncated for assumed upper-bound cutoff values (see discussion for IPPSS report Sections 7.2 and 7.9.4). The WCC curves developed in Section 7.9.2 were based on a damage-effective ground acceleration parameter and were also similarly shifted and truncated. (See discussion for IPPSS report Sections 7.2 and 7.9.2)

The results of the sensitivity analysis are presented in Table 1. The combined results for the shifted and truncated curves at the bottom (0.8×10^{-4} for Unit 2 and 1.6×10^{-6} for Unit 3) should be the same as the IPPSS results for Units 2 and 3. We believe that the difference is due to the procedures used to perform the integration and the coarseness of the hazard and fragility data points. In addition, there may be some difference due to the lumping of curves done in the IPPSS analysis (Figure 7.2-4 does not replicate the seven D&M curves from Figures 7.2-1 and 7.2-2 and the four WCC curves from Figure 7.2-3). In some sense, the difference in the results represent an analysis procedure error or uncertainty. In general, we believe that the data points for the hazard and fragility curves in the IPPSS are too coarse. A more refined set of points should be developed.

We note that the difference between the modified and unmodified results (i.e., unshifted and untruncated to shifted and truncated) are more significant for Unit 3 than Unit 2. In the integration for Unit 2 the mean value for 2RW release category is dominated almost entirely by the failure of one component with a median capacity of 0.27g, which is generally within the body of the hazard curves (which is more true for the D&M curves). In contrast, the difference is much larger for Unit 3. The median fragility capacity for this plant is approximately 0.8g which is at the upper tail of the hazard curves. For Unit 3 the results are dominated by uncertainty and depend on the upper-bound cutoff values for the hazard curves.

Several conclusions can be made based on the results of the sensitivity analysis.

1. The mean frequency of release category 2RW for Unit 2 is greater by a factor of approximately 12 between the D&M and the WCC shifted and truncated hazard curves (i.e., 1.5×10^{-4} per year compared to 1.3×10^{-5} per year). These are the curves ultimately used in the IPPSS analysis. Note that for Unit 3 the difference is about a factor of 1000. The reasonableness of this result is discussed for IPPSS report Section 7.2. Based on this study, it is clear that the WCC hazard curves are considerably different from the D&M curves.

2. For the D&M hazard curves the difference between the unshifted and untruncated results and the modified results is a factor of less than 2 for Unit 2 and slightly over 3 for Unit 3. The low factor for Unit 2 is because the median fragility value of 0.27g for Unit 2 is well away from the upper-bound cutoff values. For Unit 3 the effective median fragility value of 0.8, is at the upper limit of the cutoff values. Note that plots of the hazard curves are given in IPPSS Figures 7.2-1 through 7.2-4.

3. For the WCC hazard curves, the difference between the unshifted and untruncated results and the modified results is a factor of 13 for

Unit 2 and a factor of almost 500 for Unit 3. The high factors for both units is because the median fragility values are at or above the upper-bound cutoff values.

4. The difference between the shifted and truncated combined results (which are the basis for the final values given in the IPPSS report) for Units 2 and 3 is over two orders of magnitude. The reason is due to the effective capacity for Unit 2 being 0.27g and for Unit 3 being 0.8g.

The experience we gained in these analyses was used in estimating the effects of potential changes of individual parameters of the safety-related structures and components, and to judge the adequacy of the hazard analyses.

TABLE 1
RESULTS OF SEISMIC SENSITIVITY ANALYSIS

Category	Mean Frequency, Release Category 2RW (per year) Unit 2	Unit 3
D&M		
Unshifted and Untruncated	2.6×10^{-4}	1.1×10^{-5}
Shifted and Truncated	1.5×10^{-4}	3.2×10^{-6}
WCC		
Unshifted and Untruncated	1.7×10^{-4}	1.6×1^{-6}
Shifted and Truncated	1.3×10^{-5}	3.5×10^{-9}
Combined Results		
Unshifted and Untruncated	2.2×10^{-4}	6.2×10^{-6}
Shifted and Truncated	0.8×10^{-4}	1.6×10^{-6}
IPPSS Results	1.4×10^{-4}	2.4×10^{-6}

2. OVERALL METHODOLOGY

A general discussion of the overall methodology used to obtain the probabilistic description of failure for earthquake, flooding, and wind is presented. Specific comments on the IPPSS report sections are given in Chapter 3. The purpose of this chapter is to give general impressions of the adequacy of the procedures used.

SEISMIC

Our impression of the methodologies for seismic hazard and seismic fragility development used in the IPPSS are given below.

Seismic Hazard

The seismic hazard methodology employed in the IPPSS is appropriate and adequate to perform a seism'c risk analysis. The procedure is well established and accepted. An important element of the seismicity studies conducted for the Indian Point site is the explicit treatment of the sources of variability in the analysis. The uncertainty in the analysis can be attributed to the limited data available on eastern U.S. seismicity and ground motion. This uncertainty is reflected in the final family of seismicity curves.

The two seismicity studies performed for the IPPSS clearly identify the fact that variability due tr modeling assumptions, or uncertainty as defined in the seismic fragility a lysis, can contribute significantly to the variability in the frequency of exceedance curves. In addition, the statistical variability due to limited data and the inherent randomness of the process, which is combined with the modeling uncertainty, is also a significant contributor to the variablility in the final family of seismicity curves. A recommendation for future PRAs would be to separate these factors so they can be consistently and rationally treated in a hazard analysis. Current analysis techniques generally ignore modeling variability or treat it in an ad hoc manner.

In the last ten to fifteen years the procedures being used to conduct a probabilistic seismic hazard analysis have improved considerably and stabilized in their basic probabilistic format. However, it is equally well recognized that the analyst is allowed considerable discretion as to model selection and assumptions in application. This allowance is considerably greater in evaluating the seismic hazard in the eastern U.S. The IPPSS is an example of this.

In conducting seismic hazard analyses, seismicity data is generally collected from available catalogs. We note in the IPPSS and in most applications, that a detailed review of the data for accuracy of event sizes and locations is not made. A consequence of this has led us to conclude that inconsistencies can enter into the analysis. We are therefore of the opinion that seismic hazard analyses, particularly those relying on Modified Mercalli Intensity, should include thorough reviews of the seismicity data, and, if necessary, include an investigation of the details of critical earthquakes in the data base. The investigation should be made on the basis of state-of-the-art analysis tools and standards for assigning event size.

An alternative often chosen to model the seismic hazard in the eastern U.S. is the use of Modified Mercalli Intensity (MMI) as the parameter to characterize the size of earthquakes and the intensity of ground. This approach was adopted in the WCC seismicity study. The differences in the two seismicity studies demonstrated in Chapter 1 led us to raise questions about the overall consistency of the studies. Note that the original seismicity data is essentially the same in the two studies. We suspect that the use of intensity as a source parameter, to which a peak ground acceleration is related, leads to results whose meaning cannot be entirely known. The reason for this is the fact that intensity, by definition, is a measure of response of masonry buildings, tombstones, railroad tracks, the ground, etc. In each case of observed response, a transfer function is implicitly involved which produces the result that is observed after the event. Not surprisingly, efforts to later relate peak acceleration to intensity, exhibit considerable scatter. The reason is because information is lost concerning the transfer of seismic energy in the form of ground motion to structure response. We feel that in a way not very well understood, intensity-acceleration attenuations are values smoothed by the complex response process of those structures considered in the intensity scale. As a minimum, the careful use of this approach is indicated. On a more broad scale the need for uniformity in intensity scales is apparent.

Seismic Fragility

The methodology used in the IPPSS report for seismic effects is appropriate and adequate to obtain a rational measure of the probability distribution of the frequency of core melt and associated release categories. In the application of the methodology, we offer the following comments.

The notion of separating variability into randomness and uncertainty components is an appropriate concept. Randomness by definition is irreducible while uncertainty in the parameters and models can be eliminated by analysis, testing, research, or combinations of these techniques. However, it is our experience that in practice these definitions become blurred. What is randomness today may be uncertainty tomorrow. In other words, as the state-of-the-art advances, new techniques are developed which can be used to solve problems which yesterday were unsolvable. Even the classic example of the randomness of compressive stresses obtained from testing concrete cylinder samples may some day fall prey to an advanced analysis technique. Hence, knowing for certain the values of some obscure set of parameters (e.g., aggregate shape and location, cement properties, etc.), the compressive stress may be predicted almost perfectly. In reality, this may never occur, because today we have remaining such a small randomness component that there may not be sufficient incentive to pursue the development of a more refined theory.

In the methodology used in the IPPSS report, the median capacity value is the only uncertain parameter. It should be kept in mind that there are other uncertainties associated with the methodology (e.g., randomness β_r, the lognormal model, and even β_u itself). It is implicitly assumed that the variability in these other parts of the methodology, is relatively small so that their uncertainty can be neglected. Also, there is some evidence that variability may be constant with response level (Ref. 1*).

There are some who believe that all variability is uncertainty and the frequency of failure (fragility) curve for a component is equal to 0 up to some uncertain acceleration value and equal to 1 for higher values (i.e., the "cookie-cutter" fragility curve). Others choose to think of variability as being all randomness. The IPPSS report has taken a middle road and considers both types to be present. The implication of how dependencies are affected by these two types of variability is discussed later in this report. We personally feel that generally it is more rational to have more uncertainty and less randomness for structural components subject to seismic and other forces.

It is important that the industry adopt a consistent approach to be applied to PRA analyses. In this manner, results between PRAs can be compared (e.g., "apples with apples"). It is naive to think that the answers we produce are absolute truth. The best we can do today is to be rationally consistent and to communicate to others exactly how our analyses are performed, so that the results can be compared in a relative sense.

After reviewing the procedures used to produce the fragility data, we have a general impression which bears on the issue of consistency. We feel that the uncertainty of the parameters in the IPPSS report has probably been understated. There are various levels of sophistication which have been used to develop the fragility parameter values, but we do not sense that enough uncertainty has been assigned to components where parameter values are based on more distant information. Although in fairness to the IPPSS report, the values for β_u are generally larger for generic components as compared to plant specific components.

On the other hand, we also believe that the median capacity values are probably low. Structural and mechanical engineers have an inbred tradition to be conservative, and our guess is that this tendency has persisted in developing median capacity values. It is useful to remember that the median value is the value in which there is a 50 percent chance that the "true" value is

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*References for Chapter 2 are given at end of Chapter.

larger. We suspect that over-conservatively stating the median values and understating the uncertainties is sufficiently self-compensating such that reasonable final results are still obtained.

Several obvious elements of uncertainty have been left out of the seismic fragility analysis. First, design and construction errors (e.g., the problem of piping supports at Diablo Canyon) and aging effects are not included in the seismic fragility or fault tree analysis. These become extremely important issues for series systems such as piping systems and cables (i.e., cable trays). One failure and the system may be lost. We noted for several sections which we reviewed that the authors did not check the calculations which formed the basis for the fragility parameters that were developed. Thus, errors in the calculations could not be discovered.

In an approximate way the lower tail on the lognormal distribution for capacity accounts for possible errors. This is true since the capacity tail goes to zero which is not supported by reality. However, the frequency distribution for design and construction errors certainly varies from component to component. Since the lognormal tail is a function of only the capacity parameter, it may or may not properly account for these types of errors. Our conclusion is that design and construction errors are not specifically accounted for in the analysis.

Another uncertainty (and bias in the median value) is created by the fact that structural components are not built to produce the maximum allowable stress. Construction practices dictate that components generally are stronger than needed. It is tempting, but incorrect, to say that design and construction errors can be balanced by overconstruction such that these effects in total can be neglected. We feel these considerations individually should be taken into account in the systems analysis.

In the IPPSS report the weakest part of a structure or equipment was used to develop fragility values. In general, this approach is satisfactory. It should be pointed out that it is possible for a slightly stronger part to produce a greater frequency of failure. This occurs if the variability of the stronger component is large enough to overcompensate for the weaker but less variable part. Thus, it is not always sufficient to consider just the weakest part. Slightly stronger parts should also be reviewed and disregarded if their variability is found to be relatively small.

One approach used to develop fragility curves was based on analysis of generic data. Rather than working with the analysis of a plant specific component, failure and/or response data from similar components in similar environments are used as the basis to develop a fragility curve for the particular plant component being considered. We feel this procedure is appropriate under certain circumstances. If after determining the fragility of a particular plant component using generic data it is found that the capacity is sufficiently high so that the component does not influence the release category analysis, then we feel the analysis is appropriate. On the other hand, if the component is found to have a low capacity such that it influences (or could if changed by a small amount) the frequency of core melt analysis, then a more detailed analysis for that component should be conducted. In the IPPSS, one component in this category for the current seismic failure analysis (i.e., other components may become critical if the strength of critical components are changed) is the diesel generator fuel oil tank for Unit 3. This component, along with the control building N-S shear walls, dominate the failure frequency for Unit 3. Other components in this category are identified in the review of Section 7.9.3. We feel that because the capacities of these components are low, more detailed analyses should be conducted to verify that the generic-based capacities are appropriate.

It is important that median parameter values be selected to give frequency of behavior (i.e., failure, capacity, response, etc.) at acceleration values which are significant to the frequency of core melt analysis. In the integration of the hazard and fragility curves, the major contribution to the mean frequency of core melt will generally come from a specific range of acceleration values. For example, in developing the median factor for damping, the stress level in a structure for this range of accelerations should be taken into account in selecting the structure damping value. If the stress level is less than yield, then 3 percent may be appropriate, or if yield level is reached, 10 percent may be more representative. This is particularly important for equipment items which have natural frequencies close to a fundamental building frequency. Based on the discussion at our meeting with PLG, it is our understanding that the yield level of the structures are below the yield level for the safety-related equipment supported in the structures. Thus, it is appropriate to use damping values for a structure corresponding to the yield level.

One assumption implicit in the methodology is that everything occurs at once, and no phasing of events is considered. Structures and components either fail or do not fail at the same instant in time. For ductile structures, the loading sequence is less critical compared to the maximum load or number of cycles of large motion. For brittle elements, the loading sequence is more important. There is a dependence between the loading and response in reality, because structures fail sequentially leading to many possible failure histories. We wonder how this process might be applied to electrical control functions and the interaction of electrical equipment functional failures with failures of structural elements.

As reviewers, there is one area which is missing from the IPPSS report which should be part of all public documentations of PRA studies. Results of sensitivity calculations should be performed to provide the reader with an understanding of what elements control the results of the analysis. For example, how sensitive is the frequency of core melt to the upper-bound earthquake magnitude cut-off? What would happen to the mean frequency of core melt if the median acceleration capacity of the control room failure for Unit 2 was one-half of the computed value? As discussed in our introduction chapter, we have attempted to do this to a small degree to assist us in our review. We feel that the results of sensitivity studies should be provided as part of all basic PRA documentation.

In our review of the IPPSS report, we spot-checked calculations which could easily be done as we read the report. We also performed sensitivity studies of the hazard and fragility curve integration (see Chapter 1). In addition we reviewed the calculations for dominant components as part of our review of Section 7.9.3.

As a result of our tour of the Indian Point Site, we question whether the IPPSS has considered all possible failures of non safety-related structures or equipment, which could impact on safety-related items. The IPPSS has included, for example, possible failure of the stack, superheater building, and the turbine building onto the Unit 2 control building for seismic loads. It was pointed out during the tour that the nitrogen bottles in the Unit 3 AFW pump room could fail and the released gas propel them into safety-related control cabinets. This type of secondary failure was not considered in the analysis. Another possibility which was not documented in the IPPSS report is potential failure of the polar crane structures in the containment buildings and possible failure onto equipment below. We believe that a systematic study should be conducted to identify and quantify the effects of possible secondary failures which could affect safety-related structures and equipment.

One area which we have not commented on concerns the adequacy of the fault and event trees, except we question the absence of consideration for a moderate size earthquake occurring during a time when some safety-related components may not be available due to maintenance procedures, etc. Our understanding is that Sandia will make comments in this area. Thus, for the purposes of our review, we accept the fault trees given in the IPPSS report. In addition, Sandia has reviewed these trees and has determined that the safety-related components which are included are complete. Based on the fault trees presented in this subsection, we checked the Boolean algebra and determined that the final expressions for M_S and the various release categories are correct, except as noted.

FLOODING

The possible contribution of flood events to a core melt frequency have been evaluted in the IPPSS for external and internal flood sources. The methodology employed for external flood hazards is a departure from the analysis conducted for other external events such as seismic, hurricane and tornado. The method is somewhat ad hoc in the sense that a complete probabilistic hazard assessment was not conducted (i.e., uncertainty in key parameters are not considered, and a family of flood elevation hazard curves was not produced.) Although the state-of-the-art in flood hazard assessment is sufficiently developed to conduct such an analysis, external flooding in the IPPSS is not treated as thoroughly in a probabilistic context as other external events. An outline of a procedure to perform a flood hazard assessment is provided in Reference 2.

We do not agree with the methodology applied in the IPPSS to evaluate external flood hazards at the site. The approach provides point frequency estimates for single events and event combinations rather than considering a full complement of event sizes, parameter values, and joint occurrence of events. Therefore, at a given frequency of exceedance the uncertainty in flood depth cannot be evaluated, nor can the probability distribution on frequency. We recognize that a reason for this approach is due in part to the traditional notion of a probable maximum flood (PMF). Since the PMF is an extreme event, an annual frequency of occurrence is typically not determined by hydrologists, nor is the variability in key parameters considered. Nonetheless, the uncertainties in estimated frequencies of extreme events are generally considered to be large (Ref. 2). Similarly, for a given storm, there are important sources of uncertainty to be considered in the estimation of flood water surface profiles. The IPPSS has not conducted a sensitivity analysis nor has an analysis been conducted to obtain the uncertainty in the frequency of exceedance. In our judgment a flood hazard analysis should be conducted that accounts for modeling variability and the variability in key parameters of the flooding process.

An analysis was conducted to consider the impact of internal floods on safely related equipment and the frequency of core melt analysis. We generally agree with the steps performed in the analysis. However, the steps are not given in the IPPSS but were provided at the meeting with PLG. We recommend that the methodology and procedures applied be described in the IPPSS report. These steps follow the recommendations in Reference 2; however, there are differences in the mechanics of conducting the analysis. In future PRAs we would recommend the use of a more systematic approach, such as Reference 2 recommends.

An internal flood analysis was conducted for Unit 2. An assumption was made that similarities in the design of Units 2 and 3 allows the analysis to apply to both units. To fully accept this assumption, consideration should be given to such factors as age, temporary floodway blockages, changes in plant structure, etc. In general, we agree with the conclusions of the analysis that any flooding damage will be localized and will not result in a plant transient.

WIND

Our impression of the methodolgies for wind hazard and wind fragility used in the IPPSS is discussed in the following sections.

Wind Hazard

Extreme winds were categorized as tornados, hurricanes and extratropical cyclones and thunderstorms. Hazard curves were developed for each category. The hazard results for extratropical cyclones and thunderstorms were combined with the hazard results for hurricanes. In the IPPSS analysis it was assumed that these wind hazards are statistically independent. Comments concerning the methodology for the three wind types are given below.

Tornado: Hazard functions were developed specifically for the site. Although, hazard curves could have been developed for specific structures or group of structures, we believe that in regards to the state-of-the-art it is adequate to only develop site specific data.

In our comments for Section 7.9.5 we feel that the procedure used to determine the mean rate of tornado occurrence for the Indian Point site produced very conservative values. We believe that it is more appropriate to examine the local differences which exist in contrast to regional averaging of tornado occurrence statistics. We conclude that the approach used in the IPPSS is conservative.

We found in performing an approximate hazard analysis that reasonable differences in the wind speed, tornado length and width values, and other physical parameters did not cause a large change in the results. This gave us a sense of confidence that the distribution of tornado wind occurrence at the site is reasonable.

We do not agree with the methodology used to develop the probability distribution of hazard curves. The approach used in the IPPSS was to identify lower, median, and upper bound values for each of the basic parameters. Then three hazard analyses were conducted with the corresponding three parameter sets. The results for the lower and upper bound parameter sets were assumed to define the 5th and 95th windspeed percentiles, respectively. We believe that this approach is arbitrary and does not lead to a believable probability distribution. We recommend that a stratified statistical sampling procedure be used with multiple hazard analyses to develop a probability distribution for the tornado hazard curves.

Hurricane: We generally concur with the methodology used to develop hurricane wind speed hazard curves. The methodology rigorously considers the various basic parameters pertinent to the problem. However, because the methodology does not account for the complex site condition which exists at Indian Point or the potential wind channelization by the local hills and river valley we believe that the frequencies of occurrence are low for recurrence intervals greater than about 200 years at Indian Point.

The probability distribution of hurricane hazard curves were developed in the same manner as done for tornados. Our comments concerning the methodology used to develop the probability distribution of hazard curves for tornado are also applicable to the hurricane analysis. In summary, we do not believe the resulting hurricane probability distribution, and recommend that a more consistent and rigorous approach should be used.

Extratropical Cyclone and Thunderstorm Risk: The probability distribution of annual maximum gust speeds of fully-developed pressure system storms was approximated by the Fisher-Tippett Type I distribution. The data from the LaGuardia station, which is about 50 miles from the Indian Point site was used to develop the statistics of the wind speed distribution. As we noted in the review, new data from Reference 3 indicates that the wind speed used in the IPPSS may be low by about 10 percent. We do not feel this is potentially serious since hurricane hazard/fragility curves tend to be more important to the risk of offsite consequences.

The procedure leading to the distribution of hazard curves is based on the sampling error for the 31 observations comprising the LaGuardia data set. We feel that part of the uncertainty, which was not accounted for, is due to the fact that the Indian Point site is about 50 miles from the observation station. However, we judge that the distribution is on the conservative side since terrain roughness was not taken into account.

Wind Fragility

We believe that the methodology used to develop the fragility curves for tornado and hurricane wind speeds and tornado missile impact is not adequate. The basis for our position is given below.

The wind speed hazard curves for tornado and hurricane were based on the original design capacity and design charts given in American National Standards Institute (ANSI) Standard 58.1 (Ref. 3). The approach used to obtain lower, median, and upper bound curves was to consider variations in terrain conditions and return period wind speeds. Also included was the effects of wind speed variation with height. We believe that the wind speed return period variation is related to the hazard functions not the fragility data. More important than the variation of wind speed with height is the effect of building shape and adjacent structures on the wind loading. Our view is that variation due to these effects are extremely important and have not been properly considered in the analysis. We are particularly concerned with possible failure of the metal structures starting with the tearing of roofing or siding at a corner where the suction coefficient can be as high as 2 or 3. The only accurate way to include these spatial effects is through a boundary layer wind tunnel study of the Indian Point site. As a minimum, the very large uncertainty in the shape factors should be systematically included in the analysis. More discussion concerning this issue is given in our review of IPPSS Section 7.5.

The fragility due to tornado missiles was developed based on the conservative assumption that if a missile hits a structure, failure occurs. This implicity assumes that the missile will penetrate the structure, strike the safety-related item, and cause failure. Our problem with the methodology is that the tornado strike fragility curves were not developed considering the potential missile population and the probability of strike given a tornado at the site. Instead an argument was made based on the speed required to lift a missile off the ground. As discussed in our review of IPPSS report Section 7.5, we determine that the resulting mean frequency of impact was reasonable. However, we do not agree with the methodology leading to the development of the frequency values.

The probability distribution for wind speed and tornado strike also is not acceptable since the methodology does not properly account for the randomness and uncertainty in a systematic manner. As stated in our review of Section 7.5 the consequence analysis for wind may be low by a significant factor, and we do not believe that the confidence bounds given in the IPPSS are meaningful.

References

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2. "PRA Procedures Guide," prepared by ANS and IEEE for USNRC, NUREG/CR-2300, Review Draft, September 28, 1981.

3. New data for wind speeds at LaGuardia (will be provided on final report).

4. American National Standards Institute, Inc., "Building Code Requirements For Minimum Design Loads in Buildings and Other Structures," ANSI A58.1-1972.

3. REPORT SECTIONS

The sections of the IPPSS pertaining to the analyses for seismic, flooding, and wind were reviewed. Specific comments for the following sections are given in this chapter.

- 7.2 Seismic
- 7.4 Flooding
- 7.5 Winds and Wind Induced Missiles
- 7.9.1 Dames and Moore Seismicity Study
- 7.9.2 Woodward-Clyde Seismicity Study
- 7.9.3 Structural Mechanics Associates, Inc, Fragility Study
- 7.9.4 Structural Mechanics Associates, Inc., Damage Effective Ground Acceleration
- 7.9.5 Research Triangle Institute Extreme Wind Analysis
- 8.3.4 Identification of Major Scenarios, Systems, and Structures Contributing to Risk - Indian Point 2 (Seismic and Wind)
- 8.3.5 Identification of Major Scenarios, Systems, and Structures Contributing to Risk - Indian Point 3 (Seismic and Wind)

SECTION 7.2 SEISMIC

Scope of Review

In this section, the effects of earthquake-induced loads are reviewed. Both seismic hazard and fragility information given in this section are discussed. Additional review comments concerning the hazard curves are given for IPPSS report Sections 7.9.1, 7.9.2, and 7.9.4 and comments concerning fragility curves are given for IPPSS report Section 7.9.3. The implications of discrepancies and differences that were found are discussed. The references which were considered in the review of this section are listed below.

References

1. Aggarwal, Y. P. and L. R. Sykes, "Earthquakes, Faults, and Nuclear Power Plants in Southern New York and Northern New Jersey," Science, vol. 200, pp. 425-429.

2. Fischer, J. A., "Capability of the Ramapo Fault System," Proceedings of Earthquakes and Earthquake Engineering: the Eastern United States, September 14-16, 1981, Knoxville, Tennessee.

3. Ratcliffe, N. M., "Brittle Faults (Ramapo Fault) and Phyllonitic Ductile Shear Zones in the Basement Rocks of the Ramapo Seismic Zones New York and New Jersey, and Their Relationship to Current Seismicity," Field Studies of New Jersey Geology and Guide to Field Trips, Rutgers University, Newark, New Jersey, 1980.

4. Yang, J. P. and Y. P. Aggarwal, "Seismotectonics of Northeastern United States and Adjacent Canada," J. Geophy. Res., vol. 86, pp. 4981-4998, 1981.

5. Cornell, C. A., H. Banon, and A. F. Shakal, "Seismic Motion and Response Prediction Alternatives," Earthquake Engineering and Structural Dynamics, vol. 7, pp. 295-315, 1979.

6. McCann, M. W. and D. M. Boore, "Variability in Ground Motions: Root Mean Square Acceleration and Peak Accelerations for the 1971 San Fernando, California Earthquake," submitted to the Bull. Seims. Soc. Am., 1982.

7. Darragh, R. B. and K. W. Campbell, "Empirical Assessment of the Reduction in Free Field Ground Motion Due to the Presence of Structures," (Abstract), Earthquake Notes, 52, 1981.

8. Kennedy, R. P., et al., "Subsytem Fragility," Seismic Safety Margins Research Program, (Phrase 1), Lawrence Livermore National Laborartory, Livermore, California, NUREG/CR-2405, UCRL 15407, February 1982 (Prepared by Structural Mechanics Associates).

SECTION 7.2.1 METHODOLOGY

We agree that a seismic safety analysis consists of the five main steps which are listed in this section.

SECTION 7.2.2 SEISMICITY

This section of the IPPSS describes the seismicity studies conducted by Dames and Moore (D&M) and Woodward Clyde Consultants (WCC) and the general method used to combine the results of the two studies to provide a family of seismicity curves. The review of this section is limited to general comments about the method of analysis and the developement of the family of seismicity curves. A review of each seismicity study is presented in the comments for Sections 7.9.1 and 7.9.2 which contain the D&M and WCC studies, respectively.

In both seismicity studies, a Ramapo fault zone was not explicitly considered. However, in recent years considerable scientific study of the geology, historic and recent seismicity, have lead to a hypothesis that a Ramapo fault zone is an alternative hypothesis that should be considered in the hazard analysis (Ref. 1, 2, 3, and 4). Since the geometry of the fault zone, seismicity parameters, and a maximum event size are difficult to determine we feel that a family of seismicity curves for a Ramapo fault should be considered. The absence of the Ramapo zone from the final family of seismicity curves is, in our judgment, an inadequecy in the analysis. Chapter 4 of this report investigates the implications for release category 2RW if a Ramapo fault zone is included in the seismicity analysis. We believe that hypotheses associated with the Ramapo fault are reasonably well contained within the dispersion of the D&M family of seismicity curves.

To generate the family of seismicity curves, the results of the D&M study have been modified in two ways. First, sustained-base peak acceleration values have been shifted by a factor of 1.23 to provided sustained acceleration; and second, the hazard curves have been truncated to reflect the belief that there is a maximum ground shaking intensity which can occur. The basis for limiting peak ground acceleration given a specific value of intensity is discussed in Section 7.9.4. We offer comments here in regards to limiting the maximum intensity value.

The WCC seismicity results were not modified in this section as truncations were applied in the original study. As discussed at the PLG meeting, it was agreed that the truncation of the hazard curves should more appropriately have been performed within the probabilistic analysis. However, as pointed out at the meeting, and verified by a separate calculation, truncating outside the hazard analysis is conservative in that the annual exceedance frequencies for accelerations below a truncation level will be higher than had the truncation been performed in the probabilistic analysis.

In the D&M and WCC studies, ground shaking is a function of the intensity of the earthquake at the source and distance between the source and the site. Each study used a different method to characterize the seismic source and thus applied different models to attenuate motion to the site. Modified Mercalli Intensity (MMI) data was used in each study to develop earthquake recurrence relationships. In the D&M study MMI values were converted to body wave magnitude (M_b). Earthquake recurrence relationships and acceleration attenuation models were then described in terms of magnitude. The conversion from MMI to M_b was made through an empirical relation developed by Nuttli. WCC on the other hand used MMI directly as the source parameter. An attenuation model was developed that attenuated epicentral intensity (I_o) with distance to obtain the site intensity (I_s). The site intensity was then converted to peak ground acceleration.

The two approaches are quite common, particularly for hazard analyses conducted for the eastern U.S. The difference between the studies is the path taken to determine sustained acceleration at the site. The choice of a source parameter and ground motion prediction model affects the degree of variability in the predicted acceleration level. As discussed in Reference 5, the effect of taking a direct versus an indirect path in making spectral response predictions can increase the total uncertainty in the estimate. The study demonstrates that the total variability in the ground motion parameter is dependent on the path taken in making a ground motion prediction. In the D&M study the path used is:

$$I_o \rightarrow M_b \rightarrow A_s$$

while in the WCC study the path is:

$$I_o \rightarrow I_s \rightarrow A_p \rightarrow A_s$$

where an arrow refers to an empirical relation, and A_p is peak ground acceleration and A_s, the desired value of sustained acceleration.

In neither of the two studies was this source of variability included in the logarithmic standard deviation, $\sigma_{\ln a}$, about the attenuation equation. In addition to the variability about a given regression equation, (e.g., $I_o \rightarrow M_b$), there can also be considerable variability in the mean curve. This point was demonstrated in Section 7.9.2 by WCC for I_o - M_b relationships (see Figure 6 in Section 7.9.2). The logarithmic standard deviation value used in each study was 0.60, which is a typical value for the variability about magnitude - distance regressions for peak acceleration. The actual impact of this source of variability was not evaluated as part of the IPPSS. However, the effect of increased values of $\sigma_{\ln a}$ on the frequency of exceedance curves was evaluated and is demonstrated in Chapter 4. For an increase of 0.20 in the logarithmic standard deviations, the increase in the frequency of exceedance is within a factor of 3 for accelerations up to 0.70g.

We note that the above arguments do not impact on the selection of acceleration truncation values, either for specific values or in the manner in which the truncation is carried out. The selection of truncation values is made by arguments independent of the path taken in making acceleration predictions.

In the seismic hazard analysis, the variability in ground motion attenuation has been accounted for by a lognormal distribution with a standard deviation, $\sigma_{\ln a}$, of 0.60, a value typical of the scatter in ground motion data. Recent studies suggest that $\sigma_{\ln a}$ is in reality a composite parameter whose components include travel path, building, and local geologic effects (Refs. 6 and 7). In fact, the variability due to buildings has been identified as a function of the depth of structural embedment. In the seismic risk analysis, soil-structure interaction effects and variability in response are considered. Since free-field accelerations are specified, it may be more appropriate to account for the part of randomness (not uncertainty) in the attenuation equation due to building effects in the soil-structure interaction factor. The standard deviation corresponding to embedment effects, σ_{Bldg}, was found to be approximately 0.07, corresponding to a factor of 1.2 for data from the 1971 San Fernando earthquake (Ref. 6). In the IPPSS, soil-structure interaction does not affect the ground motion input level.

We agree that the overall seismic hazard methodology utilized by D&M and WCC is appropriate and adequate to determine frequency of exceedance curves on levels of ground shaking. Although the general probabilistic methodology is the same in both studies, there are differences in how the ground motion models were applied, the selection of key parameters, and the definition of seismic source zones. In our judgment, the Woodward Clyde study does not accurately represent the uncertainty in the earthquake process. Because of the low upper-bound intensity values used (i.e., VII and VIII) in the WCC study; we believe that the seismic hazard is better represented by the D&M study.

SECTION 7.2.3 Fragility

The methodology used to develop the fragility curves for structures and equipment is discussed in this section. We agree that this methodology is appropriate for the Indian Point Plants. The basis for accepting the methodology and specific comments concerning application of this methodology to the IPPSS study are given in Chapter 2 of this report.

We noted the statement that the factor of safety is equal to the resistance capacity divided by the response associated with the DBE. In the probabilistic analysis, dividing median values for capacity and response implicitly assumes that these parameters are independent. Due to the effects of load combinations and failure sequences this may not always be true.

SECTION 7.2.3.1 Definition of Failure

Structural failure is defined as ". . . The onset of significant structural damage, not necessarily corresponding to structure collapse." This definition may be conservative in some cases and will tend to produce higher frequency of failure estimates compared to a definition based on collapse where functional failure is not an issue. It would be more appropriate to use a median definition and add uncertainty for the definition. We agree that it is appropriate to define failure as either rupture/collapse or loss of function, whichever occurs first.

SECTION 7.2.3.2 Fragility Curve Formulation

We agree with separating variability of seismic response and structural capacity into randomness and uncertainity components.

Use of the lognormal distribution is appropriate as long as the extreme tails of the density function do not significantly influence the results of the analysis. It was found in performing the integration of the hazard and fragility curves that most of the contribution (i.e., greater than 90 percent) to the release category 2RW for Indian Point 2 was within three standard deviations from the median value for the control building/superheater building impact fragility distribution which controlled the system fragility curve for 2RW. In contrast, the contribution to release category 2RW for Indian Point 3 was generally beyond three standard deviations from the effective median value of the structure components which contribute to the mean frequency value of 2RW (i.e., the control building and diesel generator fuel oil tanks at approximately 0.8g). We believe that the results for Indian Point 3 using the lognormal distribution are conservative since the lower tail of the lognormal density function tends to be higher than other reasonable distributions which could have been used. However, as stated in Chapter 2, neglecting possible design and construction errors may over compensate the possible conservatism in using the lognormal distribution.

The shape of the upper tail of the lognormal density function does not significantly affect the ·sults, since the cumulated probability of failure is close to 1.0, and variations in tail shape do not significantly affect the integration process and the final frequency of core melt values.

The results of the fragility analysis are given in Tables 7.2-1 through 7.2-7. As noted in Chapter 1, the review concentrated on those structures and equipment which contributed significantly to the frequency of release. As discussed in the following sections, the basis for the fragility of critical structures and equipment were reviewed in detail. Other components in Tables 7.2-1 through 7.2-7 were reviewed generally (i.e., do the fragility parameter values look reasonable, and are they consistent relative to the ma:n contributing items?). For the non-key components, the possibility that they may be much weaker than calculated in the fragility analyses was considered. Specific comments on the fragility parameters for the structures and equipment are given in review of Section 7.9.3, "Structural Mechanics Associates, Inc. Fragility Study." Some general comments on Tables 7.2-1 through 7.2-7 are included in the discussion below.

We noted that the plots of the fragility curves at the end of this section (i.e., Figures 7.2-5 and 7.2-11) for identical components from Units 2 and 3 are different. We assume that this is just a plotting error.

SECTION 7.2.4 INDIAN POINT UNIT 2

SECTION 7.2.4.1 Systems and Plant Logic

It was learned at the meeting with PLG that failure of nonbearing masonry walls would affect an area out from a wall equal to a distance of one-half the wall height. This was in response to a question concerning the statement, ". . . failure would essentially be vertical collapse of the wall." We agree that blocks may fall as far as one-half the wall height.

At the meeting with PLG, we discussed the basis for the assumption that nonrecoverable failure of electrical components is about three times the value corresponding to recoverable interruptions (i.e., relay chatter or breaker trip). We also reviewed Reference 8, which discusses the basis for this assumption. Since those components do not affect the results, even at lower fragility values corresponding to relay chatter or breaker trip, this issue is not critical for Indian Point. However, we believe that any generic component which is a major contributor should be analyzed individually to obtain component-specific capacity values.

In regard to item 16 (fan cooler ductwork), we cannot judge whether the fan coolers are mechanically capable of adequately mixing the containment gases without the ductwork. If this is true, this is sufficient reason to eliminate this component from further consideration. The argument that it is improbable that all the duct risers would fail from the same earthquake may be weak. If these components are identically constructed and attached to the same portion of the building, their capacities and seismic responses may be highly correlated. If so, then the failure of one would imply the failures of others. We did not investigate the details of construction for the fan cooler ductwork.

We concur with the assumption that the gas lines which cross the plant property do not pose a significant hazard to the plant. However, we question that their median capacity is 1.4g, since these lines were probably not designed and constructed with the same high quality assurance requirements used in the design of the plant.

At the meeting with PLG, revisions to the fragility parameter values for Indian Point 2 components were made. The following changes were noted:

Symbol	Structure/Equipment	a	R	U
6	Condensate Storage Tank	1.28g	.22	.25
7	City Water Storage Tank	.25	.25	.30
8	Refueling Water Storage Tank	.70	.22	.28
21	120 VAC Transformers	1.07	n/c*	n/c
28	RCS Power-Activated Relief Valve	3.17	n/c	.61

*n/c: not changed

The only change that might have potential effects on the analysis is the city water storage tank, which originally had a median capacity of 0.83g. Pickard

Lowe, and Garrick redid the probability analysis with the lower value (i.e., 0.25g) and found no change in the results. This was reported to us at the PLG meeting. This is reasonable because failure requires that both the city water storage tank and the condensate storage tank fail. The latter component, which has a median capacity of 1.28g, dominates the results.

As stated in Chapter 1, we did not review the fault trees (IPPSS report Figures 7.2-6a through 7.2-6f) for completeness or functional relationships. Based on the trees, we did check the logic leading to the core melt and release category equations. We found that the system equations given are correct.

Because component 2 (impact between control rooms of Units 1 and 2) dominates the analysis, possible dependence between capacities and/or responses of other components does not affect the analysis results.

In the case of piping, the pipe segments are connected in series; thus, the frequency of failures for a piping system may not be conservatively represented by the frequency of the weakest component, unless the capacities and responses of all segments are individually (i.e., capacity with capacity and response with response) perfectly correlated or unless the capacity is dominated by a single weakest component. Because piping extends a relatively long distance and is supported at many places in a structure, piping response will not be perfectly correlated. Also, because different components may come from different manufacturers or material runs, capacity also is not perfectly correlated. A similar problem also exists for electric cables supported by cable trays.

This issue was discussed at the meeting with PLG. It was pointed out by SMA that the strength of piping systems usually is controlled by only one or two elements. Thus the design stress is at or near the allowable value for only a few elements. Because other elements are over designed, the issue of dependence or independence does not affect the fragility of the piping system as a whole.

At Indian Point, cable trays were not specifically designed. Generic supports were designed and allowable distances between supports specified and used in construction. It is difficult to apply the same argument to cable trays as was given for piping systems. By the time of the second session of the PLG meeting, PLG had investigated the effects of considering multiple independent cable tray supports (i.e., 10 to 15) on the frequency of release category 2RW. It was found that considering the cable tray supports to be independent had almost no effect on the results. This is reasonable considering the relatively high capacity of a single cable tray support (i.e., median equal to 1.1g) versus the controlling median fragility value of 0.27g for impact between the Unit 1 and 2 control rooms. In general, the issue of dependence should be considered for both piping systems and cable tray supports. Additional comments on this issue are given in Section 7.9.3.

SECTION 7.2.4.2 Seismic Core Melt Frequencies

We did not directly check the distribution for core melt frequency, M_S. As discussed in the next section the analysis for release category 2RW was checked. Because of the relationships between the various components in the systems equations, core melt and 2RW for seismic effects are essentially identical for Unit 2.

We believe that the mean value of 1.4 x 10^{-4} per year for the annual frequency of core melt is low by a factor of 2 because of the hazard curves. We feel that the hazard frequency of exceedance values are better represented by the values given by D&M. Since D&M and WCC hazard values were weighted equally, and since the mean frequencies of core melt based on the WCC curves are more than a factor of 10 lower, (see sensitivity analysis in Chapter 1), the IPPSS values are doubled if only the D&M hazard curves are used. As discussed in the review of Section 7.9.3, we believe that the fragility for the dominant component (i.e., impact between Units 1 and 2 control romms) is conservative. A more detailed analysis of this failure mode would probably lead to a higher median capacity.

Because of the higher level of subjective uncertainty leading to the tails of the core melt frequency density function, we do not believe the reported 90 percent confidence bounds are credible.

SECTION 7.2.4.3 Initial Assembly Leading to Release Category Frequencies

The Boolean expression for release category 2RW was checked starting with the fault trees and found to be correct. An integration using the 11 hazard curves from IPPSS report Sections 7.9.1 and 7.9.2 with the 5 fragility curves from IPPSS report Table 7.2-4 was performed using the same relative weighting as PLG and a mean frequency value of 0.8×10^{-4} per year was obtained. This compares to the value of 1.4×10^{-4} per year reported by PLG. We believe that the difference is due to differences in the integration procedures used and possibly the lumping of hazard curves into the final family used in the DPD operation. A finer discretation of the hazard and fragility points would probably reduce this difference.

The 2RW seismic sequence is the largest contributor to latent effects. It is dominated by the impact between the Unit 1 and 2 control rooms which has a median damage effective ground acceleration of only 0.27g. It is assumed that, if an earthquake large enough to fail the control room occurs, off-site power and the gas turbine will not be available. The next most significant contributor, the superheater stack falling on the control building, has a median capacity of 0.72g, which is almost larger than the upper-bound cutoff value of 0.8g used on the seismic hazard curves. Thus this component does not contribute much to the frequency of 2RW.

Based on a review of the development of the structural capacities, we believe that the mean annual frequency for 2RW equal to 1.4×10^{-4} per year is a factor of 2 low since we feel that the hazard curves given by D&M are more representative of the Indian Point site.

The sequence leading to release category Z-1Q consists entirely of failure of the containment building shear wall. Because of the relatively high capacity for this failure mode (i.e., median value equal to 1.1g) the mean frequency of failure is only 6.8×10^{-7} per year. The frequency of Z-1Q is sensitive to the upper-bound cutoff on the hazard curves. Because we feel that the D&M hazard curves are more representative, the frequency of Z-1Q is a factor of 2 low. The reason that the frequency of release for category Z-1Q is higher for Unit 2 compared to Unit 3 is due to the large soil loading on the Unit 2 containment building.

The Boolean expression for the other Boolean equations for release categories Z-1, 8A, and 8B were checked starting with the fault trees and found to be correct. These release categories do not contribute significantly to offsite effects.

SECTION 7.2.5 Indian Point Unit 3

SECTION 7.2.5.1 Systems and Plant Logic

In regard to the seismic capacities given in IPPSS report Table 7.2-7, the control building median capacity is equal to 1.20g, which is based on a shear wall failure mode. We believe that this value may be high (i.e., unconservative) for the Unit 3 control building. Part of the argument for developing the capacity for the Unit 2 control building was that impact between Unit 1 and Unit 2 control rooms will cause failure of the hung ceiling which would fall and incapacitate all operators. Based upon visual inspection of both Unit 2 and 3 control rooms, we found that both ceilings are hung by wires without special seismic bracing. We doubt that the control room ceiling for Unit 3 has a capacity equal to 1.20g damage-effective ground acceleration. The dominant components for core melt and release category 2RW are the control building shear wall and the diesel generator fuel oil tanks which together have an equivalent median capactiy of about 0.8g. We doubt that the hung ceiling in the control room has a capacity even that high.

The capacity of the diesel generator fuel oil tanks, which are buried, are based on generic data. Because this component contributes significantly to core melt and 2RW, a specific analysis for this component should be conducted.

Comments concerning the capacity of electrical components, piping and cable tray dependencies, and failure behavior of nonbearing masonary walls are the same as for Unit 2 as discussed above for IPPSS report Section 7.2.4.

At the meeting with PLG, revision to the fragility parameter values for Indian Point 3 components were made. The following changes were noted:

Symbol	Structure/Equipment	A	β_R	β_U
4	Condensate Storage Tank	1.28g	.22	.25
5	City Water Storage Tank	.25	.25	.30
6	Refueling Water Storage Tank	.70	.22	.28
27	RCS Power-Activated Relief Valve	3.17	n/c*	.61

*n/c: not changed

Since failure of both the condensate storage and city water tanks is required for loss of function, these changes will not significantly effect the results of the analysis.

As stated in Chapter 1, we did not review the fault trees (IPPSS report Figures 7.2-12a through 7.2-12f) for completeness or functional relationships. Starting with the fault trees, we did check the logic leading to the core melt equation. We found that the equation for M_s is correct.

It is doubtful that any dependence between the components will affect the analysis results. Note that perfect dependence due to ground motion is implicitly assumed in the procedure for integrating the hazard and fragility curves. Since the control building and fuel tanks are separate structures, no capacity or other response dependence is present.

SECTION 7.2.5.2 Seismic Core Melt Frequencies

We did not directly check the distribution for core melt frequency, M_s. As discussed in the next section, the analysis for release category 2RW was checked. Most of the contribution to core melt comes from 2RW.

We believe that the mean value of 2.4×10^{-6} per year for the annual frequency of core melt may be low due to potential failure of the control room ceiling and our belief that the D&M hazard curves are more representative of the Indian Point site. We feel that these differences would change the reported value by a factor of about 8. We do not believe that the reported 90 percent confidence bounds are credible.

SECTION 7.2.5.3 Initial Assembly Leading to Release Category Frequencies

The Boolean expression for release category 2RW given on IPPSS report page 7.2-20 was checked and could not be verified.

The expression that we obtained follows:

$$
\begin{aligned}
2RW = {} & \overline{32} \vee \hat{15} \vee \hat{18} \vee \Big[(24 \vee 35) \wedge (6 \vee 21 \vee 25 \vee 34) \\
& \wedge \Big\{ \Big\{ (6 \vee 27) \wedge \Big[(5 \wedge (4 \vee 0.05\ 3)) \vee 23 \vee 26 \Big] \Big\} \\
& \vee ((7 \vee 33) \wedge (6 \vee 31)) \Big\} \Big]
\end{aligned}
$$

Our understanding is that PLG used the following upper bound expression in the actual calculation.

$$2RW < \hat{32} \wedge (15 \vee 18 \vee \big[19 \wedge (11 \vee 13) \big] \vee 24 \vee 28 \vee 35)$$

We agree that this equation is a reasonable approximation; however, it is not strictly an upper bound.

An integration using the 11 hazard curves from IPPSS report Sections 7.9.1 and 7.9.2 with the 5 fragility curves from IPPSS report Table 7.2-8 was performed using the same relative weighting as PLG and a mean frequency value of 1.6×10^{-6} per year was obtained. This compares to the value of 2.4×10^{-6} reported by PLG. We believe that the difference is due to differences in the integration procedures used and possibly the lumping of hazard curves into the final family used in the DPD operation. A finer discretation of the hazard and fragility points would probably reduce this difference.

The 2RW seismic sequence is the second largest contributor to latent effects. It is dominated by the capacities of the control building shear wall and the diesel generator fuel oil tanks, which together have an equivalent median capacity of about 0.8g. We believe that the capacity of the hung ceiling in the control room may be lower and the D&M hazard curves are more representative for the Indian Point site and thus the mean frequency of a 2RW due to seismic effects is 10 times larger. Since fire dominates 2RW with a contribution of 6.1×10^{-5} annual frequency, an increase of seismic by a factor of 10 (i.e., to 2.4×10^{-5} per year) will increase the total 2RW mean frequency by a 30 percent (a small amount).

We believe that the capacity for the diesel generator fuel oil tanks should be developed based on specific rather than generic data since this component is a major contributor to seismic 2RW

The sequence leading to release category Z-1Q consists entirely of failure of the containment building shear wall. Because of the relatively high capacity of this failure mode (i.e., median value equal to 1.7g) the mean frequency of failure is only 3.7×10^{-8} per year. This result is sensitive to the upper bound cutoff on the hazard curves. Because we believe the D&M hazard curves are more representative, the frequency of Z-1Q is a factor of 2 low.

The Boolean expression for the other Boolean equations for release categories Z-1,8A, and 8B were checked starting with the fault trees and found to be correct. These release categories do not contribute significantly to offsite effects.

SECTION 7.4 Flooding

Scope of Review

In this section, the effects of external and internal flooding at the Indian Point plants are reviewed. In the IPPSS report Section 7.4.1 external flood hazards at the plant site were considered, and in Section 7.4.2 the impact of internal flooding on safety-related equipment was considered. The adequacy of these analyses are reviewed and the implications of discrepancies are discussed. The references utilized in our review of this section are listed below.

References

1. Westinghouse Electric Corporation Drawing, United Engineers and Constructors, Inc. Drawing Number, 9321-F-15353.

2. Indian Point Unit 3, PSAR, Supplement 1, Item 18.

3. Corps of Engineers, Proceedings of the American Society of Civil Engineers, Journal of Waterways and Harbors Division, Hurricane Study of New York Harbor, February 1962, Issue No. 1.

4. Quirk, Lawler and Matusky Engineers, "Evaluation of Flooding Conditions at Indian Point Nuclear Generating Unit No. 3," Revision of Report of February, 1966, April 1970.

5. Indian Point Unit 3, FSAR, Supplement 10, January 1973.

6. Quirk, Lawler and Matusky Engineers, letter to Mr. John Inglima of Consolidated Edison Company of New York, Inc., dated January 21, 1972.

7. Burkham, D. E., "Accuracy of Flood Mapping," Journal of Research of the U. S. Geological Survey, vol. 6, pp. 515-527, 1970.

8. Pickard, Lowe and Garrick letter to Mr. James F. Davis of the Power Authority of the State of New York, from Mr. Harold F. Perla of PLG, July 7, 1982.

9. "PRA Procedures Guide," prepared by ANS and IEEE for USNRC, NUREG/CR-2300, Review Draft, September 28, 1981.

10. Letter from William J. Cahill, Jr., Vice President, Consolidated Edison Company of New York, Inc., to Mr. Richard C. DeYoung, Assistant Director for Pressurized Water Reactors, Directorate of Licensing, U.S. Atomic Energy Commission, dated December, 18, 1972.

11. Letter from Carl L. Newman, Vice President, Consolidated Edison Company of New York, Inc., to Mr. George Lear, Chief, Operating Reactor Branch #3, Directorate of Licensing, U.S. Nuclear Regulatory Commission, dated January 20, 1975.

12. Letter from William J. Cahill, Jr., Vice President, Consolidated Edison Company of New York, Inc., to Mr. George Lear, Chief, Operating Reactor Branch #3, Directorate of Licensing, U.S. Nuclear Regulatory Commission, dated February 18, 1975.

13. Letter from Peter Zarakas, Vice President, Consolidated Edison Company of New York, Inc., to Mr. Steven A. Varga, Chief, Operating Reactor Branch #1, Directorate of Licensing, U.S. Nuclear Regulatory Commission, dated July 14, 1980.

14. Letter from Steven A. Varga, Chief, Operating Reactor Branch #1, Directorate of Licensing, U.S. Nuclear Regulatory Commission to Mr. John D. O'Toole, Assistant Vice President, Nuclear Affairs and Quality Assurance, Consolidated Edison Company of New York, Inc., dated December 18, 1980.

SECTION 7.4.1 External Flooding

The Indian Point plant is situated on the east bank of the Hudson River, approximately 43 miles north of New York City. The plant elevation is approximately 14.0 ft. which corresponds to the elevation of the screenwall structure for Unit 3 (Ref. 1). The plant grade is about 15 ft. The consideration of potential flooding at the site due to external flood is based principally on the flood studies conducted for the Indian Point Unit 3 operating license review, (Ref. 2, 3, 4, 5, and 6). The design basis of Unit 3 for external flooding, and thus the IPPSS, is based on the occurrence of extreme events and event combinations such as the Probable Maximum Flood (PMF), the Probable Maximum Hurricane (PMH), high tides, etc. The IPPSS concludes the contribution to the frequency of core melt due to external flood sources is extremely small. The basis of this conclusion is reviewed and the adequacy of the probabilistic methodology is discussed.

The principal basis of the external flooding analysis in this section is the work in Reference 4, and various supplements or revisions (Ref. 5, 6). The intent of these studies was to evaluate maximum water surface elevations at the site. On the basis of a review of potential sources of flooding on the Hudson River, the following events and event combinations were considered:

- Probable Maximum Precipitation (PMP), which is assumed to produce the Probable Maximum Flood (PMF)
- PMF and tidal flow
- Standard Project Flood (SPF) and Ashokan Dam Failure
- SPF and the Standard Project Hurricane (SPH) at New York Harbor
- SPF, Ashokan Dam Failure and the SPH at New York Harbor
- Probable Maximum Hurricane (PMH) and spring high tide.

The result of deterministic calculations for these events are provided in Table 1. The IPPSS estimates of the annual frequencies of occurrence of individual events in Table 1 range from 10^{-3} to 10^{-4}, while frequencies of event combinations have estimated values of 10^{-8} to 10^{-12}. The IPPSS concludes on the basis of the foregoing results that the contribution of external flooding to the annual frequency of core melt is extremely small. For this reason the study does not consider the impact of flooding on safety-related equipment or structures.

The approach taken in the IPPSS to assess the frequency of external flooding at the Indian Point site is to consider only the most extreme events (i.e. probable maximum events), and event combinations. The reason for this is apparently the fact these events were the basis of the flood design criteria used for the Indian Point site. This approach differs from a probabilistic flood hazard analysis that considers a full complement of water elevations due to a range of event sizes. The IPPSS has in effect chosen to consider for a given source of flooding one or two events and their resultant water surface elevations produced at Indian Point.

The approach taken to evaluate the chances that external flooding would effect safety-related equipment is not acceptable. We feel that the methodology employed has not adequately treated the sources of uncertainty in the analysis which may be large. Relevant examples of the uncertainty in flood routing and water surface elevation mapping including the uncertainty in flood routing procedures are presented in Reference 7. The study suggests that an average value of the one standard deviation in the estimate of water surface profiles due to riverine flooding is approximately 23 percent of the estimated flood depth. In addition, other sources of uncertainty include the frequency of occurrence model employed, the uncertainty about a derived frequency function, storm selection, etc.

At the meeting with PLG our concern that the uncertainty in the flood analysis was not taken into account was expressed. The response by PLG provided in Reference 8, does not address this issue.

It is not apparent in the analysis for flooding due to hurricanes that an occurrence at a location other than New York Harbor is considered. This approach is not consistent with the probabilistic hurricane analysis in Section 7.9.5.

We conclude from our review that the sources of external flood at the Indian Point site have been identified and adequately considered in a deterministic sense. However, in view of the potentially large uncertainties associated with the estimated frequencies and levels of floods, it has not been adequately demonstrated that the contribution to a core melt frequency can be neglected. Since the question of uncertainties have not been addressed at all, we feel that the present analysis is inappropriate.

SECTION 7.4.2 Internal Flooding

In this section the results of an analysis to consider the effects of internal flooding on safety related equipment is considered. At the PLG meeting, a summary was provided of the procedure used to identify sources of internal flooding and to determine their effect. Three steps were followed:

1. Identification of the sources of flooding.

2. Identify locations vulnerable to floods from those sources determined in 1.

3. Cause on initiating event and evaluate the impact.

We generally agree that these are the basic steps required to conduct an internal flood analysis. We would suggest that the internal flood analysis be conducted in a manner suggested in Reference 9 which recommends development of flood analysis fault trees. This would ensure that a thorough, systematic analysis of critical events and event sequences that may lead to a transient are considered. We suspect, based on references in the text, that existing fault trees have been used to some degree in the analysis. However, it is not clear that the effects of localized damage were included in the existing fault trees.

7.4.2.1 Noncategory I Systems

An analysis was undertaken to consider the impact of internal floods on the core melt frequency. The IPPSS study conducts the analysis for Indian Point Unit 2, and based on the similarities in the design of Units 2 and 3, it was assumed that conclusions reached also apply to Unit 3. This assumption is reasonable if it can also be assumed that age effects, particularly in locations where corrosion is likely, do not impact on the results. Also, since the units are not under the same ownership, it should also be verified that conditions have remained the same for both units. Since changes always take place, it is not apparent that equivalent alterations occur at the same time and in the same way in both units. Similarly, temporary blockage of flood passages will undoubtedly be different for each unit. These factors should be addressed in order to verify that the two units are the same. Unless significant changes between them are identified we judge that the difference in the contribution to plant risk will be small.

This section considers the impact of failure of Noncategory I systems on safety systems. The conclusions reached are based on extensive review by the utility and the NRC (Ref. 10, 11, 12, 13, and 14). The conclusion of the analysis is that the operation of safety systems will not be affected by flooding produced by failure of Noncategory I systems.

7.4.2.1.1 Quantification of Internal Flooding From NonCategory I Sources

7.4.2.1.1.1 Circulating Water Failure

A review of flood scenarios is presented due to a circulating water pipe failure. The situations described have been reviewed by the NRC staff. We note that flooding due to a pipe failure is considered to be self-limiting because the condensate pump motors and the 6.9kV switchgear will be flooded, resulting in reactor trip and loss of offsite power, respectively. This logic presumes that failure events can be counted on to limit the event. The basis for this should be further qualified.

Although a relatively high value for pipe failure is assumed, and no advantage is taken of operator corrective actions, consideration should also be given to potential incorrect action by the operator. Given the high value taken for a pipe failure, the effect of these factors is considered to be small.

7.4.2.1.1.2 Fire Protection System Failure

Electrical Tunnel Flooding

Conditions for flooding due to failure of the fire protection system are described. The basis of this event is reasonable; however, no information is provided regarding how the frequencies of valve and pipe failure were determined.

Diesel Generator Building Flooding

We agree with the conclusion that the frequency of diesel generator failure is negligible compared to other causes of failure. However, it is not clear that the frequency of inadvertent accuation has been considered. We judge that considering of this event will have a relatively small effect on the frequency of diesel generator failure.

Charcoal Filter Flooding

We agree with the conclusion and have no additional comment.

7.4.2.2 Category I Systems

7.4.2.2.1 Primary Auxiliary Building (PAS)

The analysis of flooding in the PAB has been conducted in a manner that identifies the effect of flooding due to the RWST, the service water system and component cooling system. For each system the frequency of failure has been quantified and considered in the system fault trees. These frequencies are not quantified in this section. The approach taken in this section is to identify the events that would occur in the event a flood were to occur. Since the review of the system fault trees is not a part of this review section and is being conducted by Sandia, it is assumed that the failure of the RWST, service water system and component cooling system has been taken into account. It is not apparent from the discussion that the impact of flooding was included in the system fault trees.

7.4.2.2.2 Diesel Generator Building (DGB)

Flooding in the DGB can be contained by the pit areas and the 12" drain-lines which drain to the circulating water discharge tunnel. Since a plant transient does not occur due to the diesel generators failing, the only event of interest is the joint occurrence of this event and a plant transient. The frequency of this event has been treated in the failure of the service water system. We agree with the conclusion that the likelihood of this event is small.

7.4.2.2.3 Auxiliary Feed Pump Building (AFPB)

The AFPB has been designed to discharge water from a feedwater line break. However, flood discharge rates of a feedwater line break and drainage capacities are not quantified and, therefore, this statement cannot be evaluated. Our review of this section and Reference 14 and verification that the appropriate failure frequencies are quantified for the auxiliary feedwater system, we have no further comment on this section.

7.4.2.2.4 Control Building (CB)

Flooding in the CB due to a service water break is considered. Of vital importance is the 480V switchgear located at level 15'. The analysis assumes that floor drains in the CB will remain available in the case of a flood. To fully demonstrate this, the location of floor drains with respect to the service water lines and the 480V switchgear must be provided. The conclusion is made that the frequency of power loss is less than the frequency of loss due to other causes. It should be demonstrated in the IPPSS report that the additional increase in the loss of power is negligible.

7.4.2.2.5 Containment Building

The recent experience of flooding in the containment building has led to significant changes in both units. The numerous changes which have been made are listed in the report. No quantification has been made of the frequency of flooding and damage in the containment for the upgraded facility. The reason for this is apparently that a service water system rupture and a LOCA must occur, in order to contribute to plant risk. Due to past experience, a quantification of the system reliability is called for, such as a comparison between the upgraded plant and the system at the time of the 1980 accident. We, in general, agree that the changes have increased the system reliability and that the contribution to plant risk is less than the original design.

SECTION 7.5. WINDS AND WIND INDUCED MISSILES

Scope of Review

In this section, the effects of tornado and hurricane wind pressure and tornado missle loads are reviewed. The hazard curve information is reviewed for IPPSS report Section 7.9.5. The fragility curves are given in IPPSS report Section 7.5 and are reviewed below. The implications of discrepancies that were found on core melt and release categories 2RW and 8A are discussed. The references which were considered in the review of this section are listed below.

References

1. United Engineers and Constructors, Inc., "Indian Point Generating Station - Unit No. 2, Report - Plant Capability to Withstand Tornadoes," January 26, 1968.

2. Twisdale, L. A., W. L. Dunn, and J. Cho, "Tornado Missile Simulation and Risk Analysis," Meeting on Probabilistic Analysis of Nuclear Safety, ANS, Newport Beach, May 1978.

3. Twisdale, L. A., et al., "Tornado Missile Risk Analysis, prepared for Electric Power Research Institute, EPRI NP-768, May 1978.

4. American National Standards Institute, Inc., "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," ANSI A58.1-1972.

5. Structural Mechanics Associates letter to Pickard, Lowe and Garrick, Inc., dated August 5, 1981.

6. Batts, M. E., et al, "Hurricane Wind Speeds in the United States," NBS Building Science Series 124, National Bureau of Standards, May 1980.

SECTION 7.5.1 WIND EVENTS

Review of Research Triangle Institute's wind hazard analysis report is discussed under IPPSS report Section 7.9.5. We note that wind exceedance functions were not provided for specific structures as stated in the IPPSS but rather were provided only for the Indian Point site. Discussion concerning implications of this fact are given below.

We concur with the procedure to develop hazard curves for extreme winds, hurricanes, and tornadoes separately, and to assume the results from the three sources are independent. We believe that correction factors for the effects of height, which were included in the analysis, are small relative to the influence of adjacent structures, which were not explicitly included in the analysis. This concern is discussed further for IPPSS report Section 7.5.3.

We believe that the tornado hazard curves are conservative, but that the hurricane hazard curves are unconservative. The implications of this result are discussed below for IPPSS, Sections 7.5.4 and 7.5.5.

SECTION 7.5.2 TORNADO MISSILES AND WINDS ON CONCRETE STRUCTURES

The statement that the concrete stuctures were designed for 25 psf wind loading, and that there is "little deflection" is misleading and not pertinent to the conclusion that potential wind pressures and tornado missiles are not significant to Indian Point safety-related concrete structures (i.e., wall thickness greater than 12 inches). We concur with this conclusion based on review of References 1, 2, and 3. In addition, as discussed for IPPSS report Section 7.9.5, we believe that the hazard due to tornadoes is lower than stated. The comment that the 12-to-14 inch thick walls have weights over a 150 pounds per square foot should be clarified (although true, the reviewer expected "pounds per cubic foot").

The statement that tornado frequencies at Indian Point are lower should be documented (although we do agree with this statement). In general, other leading statements made in this section should be documented.

SECTION 7.5.3 TORNADO MISSILES AND WINDS ON METAL STRUCTURES

We agree that it is conservative to base the fragility of metal structures and exposed equipment on the hit frequency; however, the fragility curves for the effects of tornado missiles were not developed based on possible hit frequencies as stated, but rather on wind velocities which could lift various missiles off the ground. However, we believe that using the tornado impact fragility curves shown in IPPSS report Figure 7.5-3 results in conservative frequencies of failure for the structures and equipment considered.

We developed our basis for this conclusion using References 2 and 3 which reported the probability of hit frequency of specific structures at a nuclear power plant. In one analysis reported, 5000 missiles which included over 2000 missiles with a mean weight less than 105 pounds were located close to the plant. For a tornado occurrence rate corresponding to USNRC tornado Region I, the mean hit frequency ranged between 1.38×10^{-6} per year to 3.09×10^{-5} per year. Adjusting for a lower tornado occurrence rate at the Indian Point site (i.e., from $4 \times 10^{-4}/yr/mi^2$ to $2 \times 10^{-4}/yr/mi^2$ or even lower as discussed for Section 7.9.5) and the size of the critical safety-related structures at Indian Point (i.e., service water pumps and RWST), a conservative hit frequency of 10^{-6} per year is obtained. From the IPPSS analysis the mean hit frequency is inferred to be 9.2×10^{-7} per year (based on release category 2RW for Unit 3 which is dominated by failure of the service water pumps). This coupled with the additional conservatism that a missile hit does not always mean failure leads us to conclude that the missile impact calculations are conservative.

We feel that hurricane-caused missiles are probably not a problem; however, this potential cause of failure should be considered and documented in the IPPSS report.

We believe that the major uncertainty in wind loading on an Indian Point structure (conditional on the occurrence of free field wind velocity) is due primarily to the influence of nearby structures. We do not believe that the randomness or uncertainty included for the capacity due to wind have been rationally developed to include the influence of the close proximity of adjacent Indian Point structures. Also, we disagree with the development of the wind load correction factor SF_L.

For hurricane winds, SF_L randomness was based on consideration of differences in terrain and return period occurrence wind speeds. The influence of nearby structures is more significant than terrain variability and should have been explicitly included. Also differences in occurrence rate belongs in the wind speed hazard analysis rather than the fragility formulation. For tornadoes, SF_L, randomness was based on the relatively insignificant differences in wind speed effects over the height of the structures. We disagree with the statement that site exposure considerations are not particularly applicable to tornado phenomena. This may be true for residential areas where tornadoes will completely destroy and flatten all structures in their path. However, at Indian Point all of the major concrete structures will survive a tornado strike. Thus, the presence of these structures will effect the flow of wind around the metal buildings and hence effect the loading on these structures.

Because of the approach used to develop the factor SF_L, the slope of the fragility curves for tornado effects are steep while the corresponding curves for hurricanes are less steep. We believe that the randomness (which is expressed by the slope of the fragility curves) should be essentially the same for the effects of tornado and hurricane wind speeds. This would be consistent with the implicit assumption made in the IPPSS that the wind speeds for tornadoes and hurricanes are the same at 33 feet above the ground. If this is true (and we believe this is a reasonable assumption), it should also be true for other elevations between the ground and top of structures at Indian Point. Implications of the slopes of the fragility curves are discussed for IPPSS for report Sections 7.2.4 and 7.2.5.

We noted two discrepancies in the development of the fragility curves. In Table 7.5-1, the velocity pressure for exposure C for a 100-year return period from Reference 4 should have been 27 psf instead of the value of 18.5 psf used in the analysis. The effect of this error would be to increase the randomness for hurricane wind fragility curves which would lead to a slightly larger frequency of core melt (probably a small effect). The second discrepancy is the conversion of pressure to equivalent wind velocity using the equation: q =

$0.00256V^2$. This equation ignores the differences between structure shapes. For example, a rectangular building in the open is more closely modeled by the equation of $q^* = 1.3q$ where 1.3 is the shape factor. Because of the influence of adjacent buildings, the shape factor will vary from structure to structure. We believe that the only rational way to develop shape factors for buildings at Indian Point is through use of a wind tunnel model. Our judgment is that the shape factors for the Unit 2 control building, the Unit 2 diesel generation building, and RWST also vary depending on the type of failure being considered. As discussed below, these structures control the core melt and release frequency analysis.

The fragility curves for the effects of wind correspond to failure of a major structural element such as the shear walls or siding. However, the local shape factor for failure at a building corner may be as high as 3.0 (negative pressure). Tearing of siding or roofing due to negative pressures is a common failure mode for metal buildings.

Assuming a local failure may control the capacity of the diesel generator building, the median capacity may be smaller by a factor of as much as 1.7; however, this building is shielded to some extent. For the RWST we believe that the implicitly-assumed shape factor of 1.0 is appropriate. Because of the location of the control building, which is relatively sheltered, the shape factor is probably 1.0 or less. However, this should be confirmed by PLG and documented.

The offsite power fragility is assumed in the IPPSS to be controlled by the fragility of the transmission line towers. Because the offsite towers have not been specifically identified and analyzed, we believe that a median fragility wind velocity value of 140 mph is unconservative. It is likely that offsite power will be lost at a much lower wind velocity. We believe that it would be prudent to assume that offsite power is not available if either a tornado or hurricane occurs. The implication of this assumption is discussed below.

We feel that there is no rational basis for the assumption that the upper-bound and lower-bound fragility curves are each weighted with probability 0.1. The result of this assignment causes the three middle fragility curves used for the hurricane and tornado analysis (see IPPSS report Tables 7.5-4, 7.5-5, and 7.5-6) to be nearly identical. Because of the apparently arbitrary assignment of probability values (i.e., 0.2 could have equally been used for the upper- and lower-bound curves), we do not have confidence in the spread of the probability distribution. Also, the mean values will change significantly for hurricanes as the probability assignments are altered. This is due to the relative steepness of the hurricane hazard curves. We have not investigated further the influence of this effect.

We reviewed Reference 5 and concur with the conclusion that the capacity of the main steam and feedwater lines correspond to an extremely high wind velocity value.

We believe that the possibility of either the turbine building or the superheater building failing and falling on the control building should be considered. Also the possibility of the superheater building failing and falling on the diesel generator building and the condensate storage tank should be considered. The fragility curves for these structures should be developed to determine whether they effect the probability of core melt and subsequent release.

SECTION 7.5.4 INDIAN POINT UNIT 2

SECTION 7.5.4.1 Plant Logic

Based on the fault trees given in IPPSS report, Figures 7.5-6 a through f, the Boolean equations leading to core melt, M_W were checked. We generally agree with the final expression given on page 7.5-12. We believe that part of the probability of the stack failing and falling on either the control building or the diesel generator building was omitted. This contribution amounts to $0.05\ 7_W \vee 0.05\ 7_T$. Because of the high capacity of the stack relative to the control and diesel generator buildings, this discrepancy has no significant impact.

The significant contributors to core melt are due to wind pressure failure of offsite power 2_W, the control building 4_W, and the diesel generator building 6_W. Note that the subscript "W" refers to either hurricane or tornado winds, while "T" refers only to tornado missile effects. The significant portion of the core melt Boolean equation is $2_W \wedge (4_W \vee 6_W)$. The other parts of the equation are not important since the capacity for tornado missiles is relatively high. The implications of the differences between our opinion and the IPPSS approach in developing the hazard and fragility curves are discussed below in connection with release category 2RW.

SECTION 7.5.4.2 Wind Core Melt Frequencies

Based on the discussion below for release category effects, we believe that the mean annual frequency of core melt value of 4.3×10^{-5} per year may be low by a factor of about 13 (a large effect). We do not believe that the confidence bounds given are meaningful.

SECTION 7.5.4.3 Initial Assembly Leading to Release Category Frequencies

Based on the fault trees given in IPPSS report Figure 7.5-6, 7.5-8, and 7.5-9, the Boolean equations leading to the release categories 2RW and 8A were checked. Implication of differences between our opinion and the IPPSS approach in developing the hazard and fragility curves is discussed for each category.

Release Category 2RW

For hurricane winds, release category 2RW is dominated by the Boolean expression $2_W \wedge (4_W \vee 6_W)$ where the symbols correspond to offsite power, the control building (which houses the switchgear and batteries for starting the diesel generator), and the diesel generator building, respectively. Other parts of the equation are controlled by tornado missile capacities which are not possible for hurricanes. As discussed for Section 7.5.3, we believe that offsite power should be considered to have failed if a hurricane occurs. Loss of AC power results in a small break loss of coolant (pump seal LOCA) sequence with no injection and no containment safeguards. Because of the steepness of the hurricane hazard curves, assuming that offsite power is unavailable, will increase the mean frequency of 2RW by a factor of at least 2. We also believe that the fragility curves may be on the unconservative side; however, due to the protection provided by adjacent structures, the implicity assumed shape factor value of 1.0 may have resulted in over predicting the control room fragility capacity for wind pressure effects.

Based on review of Section 7.9.5, we believe that the median hurricane hazard curve is unconservative. A comparison of the IPPSS median and upper-bound curves with the curve obtained from Reference 6 are shown in Appendix B, Figure 1. Using a range of hazard curves based on Reference 6 and the median fragility curve from IPPSS Table 7.5-4, we obtain a factor of 10 to 30 increase in release category 2RW. We believe that a factor of 10 increase is appropriate for differences due to the hazard curves.

In developing the Boolean equation for 2RW, part of the probability of the stack failing and falling on the control or diesel generator buildings was omitted. The capacity of the stack is relatively high and the omission of the stack failing does not significantly effect the frequency of 2RW.

In summary, we believe that the 2RW mean failure frequency value of 2.7×10^{-5} per year for hurricane effects may be low by a factor of 20 due to revised fragility for offsite power and an increase in the hurricane hazard at the site.

For tornado winds, release category 2RW is dominated by the same Boolean expression as discussed above for hurricanes. Other parts of the sequence equation (i.e., including service water pumps and the RWST) are controlled by tornado missile capacities which are high relative to wind pressure capacities. Assuming that offsite power is not available will not change the tornado 2RW frequencies quite as much as for hurricane effects. Because the hazard curves for tornado are less steep than the hurricane curves, it is estimated that, if offsite power is unavailable, the mean value will change by a factor of less than 2. We believe that the tornado hazard curves are conservative and, if decreased based on comments made for Section 7.9.5, would lower the mean value by a factor of 2 to 10. In summary, the mean value of 1.6×10^{-5} per year is reasonable and probably conservative.

Release Category 8A

The Boolean equation for release category 8A was checked, and we agree with the final results except for the small contribution from failure of the stack which was neglected. For hurricane effects, the Boolean equation leads to a nearly impossible sequence involving failure due to missiles. For tornado effects, the PAB must not fail while the RWST fails due to tornado missile. Since the capacity of the RWST is much higher than the PAB, this sequence is not very likely; thus, the probability of 8A is essentially zero.

SECTION 7.5.5 INDIAN POINT UNIT 3

SECTION 7.5.5.1 Plant Logic

Based on the fault trees given in the IPPSS report, Figures 7.5-11 a through e, the Boolean equations leading to core melt, M_W were checked. We agreed with the equations given in the IPPSS report.

The significant contributions to core melt are due to failure of either the RWST, 9_T, or the service water pumps 11_T. Other components in the sequence, such as offsite power and the AFW pump building, will fail due to wind pressure at much lower wind velocities than missile failure of the RWST or the service water pumps.

SECTION 7.5.5.2 Wind Core Melt Frequencies

Based on the discussion below for release category effects, we believe that the mean annual frequency of core melt value of 1.3×10^{-6} per year is reasonable. We do not feel that the confidence bounds given are meaningful.

SECTION 7.5.5.3 Initial Assembly Leading to Release Cateogry Frequencies

Based on the fault trees given in IPPSS report, Figures 7.5-11, 7.5-13, and 7.5-14, the Boolean equations leading to the release categories 2RW and 8A were checked and found to be correct.

Release Category 2RW

The category 2RW sequence is dominated by the failure of the service water pumps, 11_T, since failure of offsite power will occur at a much lower wind velocity. Because the RWST, 9_T, is in series with offsite power, it is not a major contributor to 2RW release. We disagree with the statement in the IPPSS report, page 7.5-19, that the auxiliary feed pump building is a dominant contributor to release category 2RW. This component is not part of the final Boolean expression.

Since missiles from hurricanes are not a significant threat and hurricane wind pressures will not fail the concrete structures, there is no contribution to 2RW from hurricanes. As discussed in review of Section 7.5.3, we believe that the failure of the service water pumps due to tornado effects is approximately 10^{-6} per year. Thus, the mean value of 9.2×10^{-7} per year for category 2RW due to wind loading is reasonable.

Release Category 8A

Since missile failure of the RWST while the service water pumps remain operable is required for a category 8A release, hurricane wind pressures do not contribute to this release category. Without failure of the fan coolers, the dominant sequence for an 8A category release is non-failure of the service water pumps, 11_T, and failure of the RWST, 9_T. Both events are associated with missile capacities. An approximate check confirmed that the mean value of cateogry 8A equal to 4.1×10^{-7} per year is reasonable.

SECTION 7.9.1 DAMES AND MOORE SEISMICITY STUDY

Scope of Review

In this section the seismicity study performed by Dames and Moore (D&M) is reviewed. The methodology used in the study to obtain a rational measure of the probability of frequency of levels of ground shaking is reviewed for adequacy and appropriateness. Important model assumptions, parameter selections, and the evaluation of significant sources of uncertainty are also reviewed. In conducting our review, the references listed below were used.

References

1. TERA Corporation, "Seismic Hazard Analysis - Solicitation of Expert Opinion," Lawrence Livermore Laboratory, NUREG/CR-1582, UCRL-53030, 1979.

2. Aggarwal, Y. P., and L. R. Sykes, "Earthquakes, Faults, and Nuclear Power Plants in Southern New York and Northern New Jersey," Science, vol. 200, pp. 425-429, 1978.

3. Ratcliffe, N. M., "Brittle Faults (Ramapo Fault) and Phyllonitic Ductile Shear Zones in the Basement Rocks of the Ramapo Seismic Zones New York and New Jersey, and Their Relationship to Current Seismicity," paper, Field Studies of New Jersey Geology and Guide to Field Trips, Rutgers University, Newark, New Jersey, 1980.

4. Yang, J. P. and Y. P. Aggarwal, "Seismotectonics of Northeastern United States and Adjacent Canada," J. Geophy, Res., vol. 86, pp 4981-4998, 1981.

Seismic Hazard Model

The seismic hazard methodology used in this study is adequate and appropriate for use in evaluating the seismic hazard. The seismic hazard model is typical of the modeling technique generally used and is a relatively stable procedure. We agree that the steps in the hazard analysis are:

- Defining of seismogenic zones
- Estimation of seismicity parameters
- Selection of an attenuation model

We note that these steps are iterated upon to consider different interpretations of the data and variations in modeling assumptions. Each step is reviewed below.

Seismogenic Zones

The selection of seismogenic zones was based principally on the work in Reference 1. Two zones are considered in the analysis: a Northeast tectonic zone and the Piedmont and Piedmont-Cape Ann zones. The Northeast tectonic zone was derived on the basis of geologic considerations and the identification of small tectonic zones. The Piedmont zone was the preferred choice of the experts polled in the TERA study (Ref. 1). The Piedmont-Cape Ann zone is an extension of the Piedmont zone to the north to include the Cape Ann area. As noted in the report, each of these source zone selections represents a rather broad interpretation of the seismicity in the region near the Indian Point site. We imply from the text, and the source zones selected, that no effort was made to review the seismicity in the region near the site.

The report addresses the issue of a Ramapo fault zone as described in Reference 2. The study concludes, on the basis of the opinion expressed by the experts in the TERA study and the conclusion reached by the Advisory Committee on Reactor Safeguards, that insufficient evidence exists to consider the Ramapo fault as an active earthquake generating source. Therefore, the source zone hypothesis set neglects a Ramapo fault zone.

Although it is difficult to access the exact spatial extent of a Ramapo fault zone and to define seismicity parameters, it is our judgment that a set of source hypotheses that does not consider a Ramapo fault zone is incomplete. For this reason we judge that the selection of source zones is inadequate in that it does not fully represent all reasonable source zone possibilities. This opinion is based on the view that recent scientific investigation of the Ramapo fault and the hypothesis that it may be a source of earthquakes in the region warrants its consideration in the analysis with some (possibly small) probability weight (Ref. 2, 3, and 4). In Chapter 4 of our report we consider the impact of including a Ramapo zone in the hazard analysis.

Seismicity Parameters

Body wave magnitude (M_b) was selected as the earthquake source parameter. To determine a recurrence relationship on M_b, historical Modified Mercalli Intensities were converted to magnitude using an empirical relation developed by Nuttli from central U.S. data. Although the relation was apparently checked with northeastern U.S. data, and verified as to its appropriateness for use in this region, this was not documented. This transformation is a source of uncertainty in the analysis. The WCC seismicity study in Section 7.9.2 demonstrated the significant effect different mean I_0 - M_b curves can have on the annual frequency of exceedance curves.

The treatment of the uncertainty in the Richter b-value is considered adequate. Also, the selection of M_b = 4.0 as the lower-bound magnitude is reasonable.

The selection of maximum magnitudes is based on the maximum observed intensity and the I_0 - M_b relationship of Nuttli. The method used to define $M_{b\ max}$ is reasonable; however, the effect of using other reasonable I_0 - M_b relations should be considered. It is anticipated that the effect of considering other relations would result in lower frequencies of occurrences.

Estimation of Seismic Ground Motion

The attenuation relation developed by Nuttli for sustained acceleration defined as a function of magnitude and distance is used. In our review of this section, we do not comment on the use of sustained acceleration as a measure of effective acceleration. Our comments on this topic are reserved for our review of Section 7.9.4. We agree that the Nuttli attenuation is a reasonable choice. The modification of sustained acceleration used in the D&M study to obtain peak acceleration is disregarded in our review because this effect is later removed when the family of seismicity is developed in Section 7.2.

The study considers the development of a peak acceleration attenuation relation by attenuating epicentral intensities, applying an acceleration-intensity relation and then converting intensity to magnitude. This alternative is rejected, (given a probability weight of zero), due to the fact that the data base used to develop the relation, is limited, and not necessarily appropriate to apply in the northeast. We note that D&M has also given a probability weight of zero to the alternative of using an acceleration attenuation function that describes peak acceleration in terms of epicentral intensity and distance.

The uncertainty about the attenuation curve is described by a lognormal distribution with a logarithmic standard deviation of 0.60. This value is typical of the scatter in strong motion data. However, since sustained acceleration is used in the analysis, it would have been more appropriate to use the logarithmic standard deviation derived in Nuttli's study. It is anticipated that this difference will be small. Also, alternate assumptions on $\sigma_{\ln a}$ could have been tested (with appropriate weights attached), but we suspect this would not have had much impact on the final results.

Results of Analysis

A series of results are presented indicating the sensitivity of the frequency of exceedance curves to variations in key parameters. The results are particularly sensitive to $M_{b\ max}$ values and the activity rate for each zone. These two factors appear to be the dominant reasons for the Piedmont-Cape Ann zone producing the highest seismicity curve. Given the modeling assumptions made in the study, the sensitivity analysis and the assigning of probabability weights to key parameters was reasonably thorough and representative of the uncertainty in the process.

Summary Comments

The seismic hazard analysis conducted is judged to be reasonably comprehensive in its treatment of the key elements of the process. A major drawback of the study is the absence of any detailed study and direct consideration of the seismicity in the area near the site. In particular, a small source zone hypothesis consisting of the Ramapo fault was not considered (i.e., it was given a zero probability weight). This is judged to be a deficiency in the present study. However, as a result of our seismic hazard analysis presented in Chapter 4, we believe that results for a Ramapo fault zone are reasonably well contained in the D&M seismicity curves.

SECTION 7.9.2 WOODWARD CLYDE SEISMICITY STUDY

Scope of Review

In this section the seismicity study performed by Woodward Clyde Consultants (WCC) is reviewed. The methodology used in the study to obtain a rational measure of the probability of frequency of levels of ground shaking is reviewed for adequacy and appropriateness. Important model assumptions, parameter selections and the evaluation of significant sources of uncertainty are also reviewed. In conducting our review, the references listed below were used.

References

1. Aggarwal, Y. P., and L. R. Sykes, "Earthquakes, Faults, and Nuclear Power Plants in Southern New York and Northern New Jersey," Science, vol. 200, pp. 425-428, 1978.

2. Ratcliffe, N. M., "Brittle Faults (Ramapo Fault) and Phyllonitic Ductile Shear Zones in the Basement Rocks of the Ramapo Seismic Zones New York and New Jersey, and Their Relationship to Current Seismicity," paper, Field Studies of New Jersey Geology and Guide to Field Trips, Rutgers University, Newark, New Jersey, 1980.

3. Yang, J. P. and Y. P. Aggarwal, "Seismotectonics of Northeastern United States and Adjacent Canada," J. Geophy, Res., vol. 86, pp 4981-4998, 1981.

DESCRIPTION OF THE SEISMIC EXPOSURE MODEL

The seismic hazard methodology used in this study is adequate and appropriate for use in evaluating the seismic hazard. The seismic hazard model is typical of the modeling technique generally used and is a relatively stable procedure. We agree that the steps in the hazard analysis are:

- o Identification of seismicity sources
- o Characterization of activity of seismicity sources
- o Characterization of attenuation of ground motion.

We note that these steps are iterated upon to consider different interpretations of the data and variations in modeling assumptions. Each step in the analysis is reviewed below.

Identification of Seismicity Sources

We agree that seismic activity beyond a 200 km radius from the site will cause negligible ground motion at the site and can therefore be neglected in the analysis.

Characterization of Activity of Seismicity Sources

We agree that the mean activity rate of earthquakes can be expressed by the Gutenburg-Richter recurrence relationship. We further agree that upper and lower bounds may be used, however, we suggest that the use of the term, "maximum credible earthquake," is an inappropriate description of the upper bound on earthquake size.

Characterization of Attenuation of Ground Motion

We agree that the two sets of attenuation equations can be used in the analysis. No further comment on this section is required.

SOURCE AREAS OR SEISMOGENIC ZONES

The selection of seismic source zones was based on the following criteria:

1. Seismic activity throughout the area appears uniform,

2. The contemporary tectonic environment and geological structures are similar throughout the area supporting the model criteria of uniform likelihood of earthquake occurrence.

Selecting source areas on this basis, five source zones were identified for consideration in the analysis. We note that among the zones considered is a comparatively small source area that encompasses the Ramapo fault. However, Source 1 zone is considerably larger than the zone proposed in Reference 1.

The WCC report addresses the issue of the Ramapo fault as a potential source zone. A conclusion is reached, on the basis of a review of Reference 1 and historic seismicity that a small Ramapo fault zone cannot be justified. Although it is difficult to access the exact spatial extent of a Ramapo fault zone and to estimate seismicity parameters, it is our judgment that a set of source hypotheses that does not consider a Ramapo zone is incomplete. For this reason we judge that the selection of source zones is not adequate in that it does not fully represent all reasonable source possibilities. We express the opinion that recent scientific investigation of the Ramapo fault warrants its consideration in the analysis with some (possibly small) probability weight assigned. Chapter 4 of this report will investigate the impact of including a Ramapo fault zone in the hazard analysis.

The WCC study has selected Modified Mercalli Intensity as a measure of the intensity of earthquakes at the source. The selection of MMI is a common practice, particularly for seismic hazard studies in the eastern U.S. This approach is an acceptable modeling alternative; however, as expressed in our review of Section 7.2, careful use should be made of this parameter in order to avoid potential inconsistencies in the analysis.

UPPERBOUND

A key parameter in the seismic hazard analysis is the choice of an upper-bound epicentral intensity. The study has chosen MMI VII with a 0.80 probability weight and MMI VIII with a 0.20 probability weight as estimates of an upper bound on epicentral intensity. From relationships between I and M_b developed in the WCC study, an intensity VII corresponds to an M_b of 4.73 while an MMI of VIII is equivalent to an M_b of 5.43. Our impression is these values are low. A comparison between the preferred WCC I-M_b relation and Nuttli's relationship are shown in Figure 1. In Appendix A to this report magnitude estimates are provided, on the basis of a cursory review of available references, of a number of the larger events that have occurred in the region. The December 19, 1937, event is estimated on the basis of the felt area, to have an M_{bLg} magnitude of 4.8 ($\pm$0.30). A similar brief review of the August 10, 1884 event suggests an M_b of 5.7.

The discussion in this IPPSS report section regarding the selection of an upper-bound intensity contains the following statement: "The composite value is not an accurate representation of our uncertainty regarding upper bound." We find this statement confusing, leaving us to question what is uncertainty on the upper bound.

We find a second aspect of this discussion to be confusing. It relates to the statement that since no "historical events of significant size have localized near the site area," that the maximum intensity is somehow limited by this fact. This implication contrasts with the notion of uniform seismicity in the source zone and is a weak basis to define a maximum event size.

For Source 1, the fact that a number of intensity VII events have occurred there, and the assigning of a probability weight of 0.80 to intensity VII expresses a degree of belief that the maximum event which has been observed on a number of occasions is probably the largest event that can be generated in the region. Figure 2 shows the location the largest events in the area near the Indian Point site.

On the basis of the above points, we judge that the assessment of maximum epicentral intensities has not been carried out consistently. We suspect that the mean value of the distribution, as well as the uncertainty in the estimate of this event are not adequately represented. As noted in other sections of this report, the selection of an upper-bound event is critical for the estimation of the frequency of exceedance for acceleration and also for the frequency of core melt and offsite consequences.

Intensity Attenuation

To express the attenuation of ground motion a model is developed in two steps; site intensity, I_s, is expressed in terms of epicentral intensity, I_0, and distance; and a peak acceleration intensity relation. The uncertainty about the attentuation relation is described by a lognormal distribution with a logarithmic standard deviation of 0.60. The standard deviation of 0.60 is claimed as being in the upper range of values used in previous studies; which previous studies is not clear. Also, since the basis of selecting the 0.60 value is not presented, we assume its selection is based on the results of regression studies on peak ground acceleration. If this is the case, the value of 0.60 is a typical value and not in the upper ranges as claimed.

RELATIONSHIP BETWEEN EARTHQUAKES AND GROUND MOTION

The intensity-acceleration relation of Trifunac-Brady was selected. It is interesting to note that the work by Murphy and O'Brien is considered to be more thorough, but is not the preferred choice. As results later indicate, the Trifunac-Brady relation is conservative in that higher frequencies of exceedance are obtained.

Intensity - Maginitude - Ground Motion

The differences in intensity-magnitude relations are discussed. The statement is made, and we feel correctly so, that there is no physical reason to expect an exact relationship between intensity and magnitude. However, as a result of the differences between the WCC and D&M analysis, caution must be exercised in conducting a hazard analysis based on intensities. We feel that it is informative when conducting such an analysis to make a comparison with I-M_b relations and to assess magnitudes of the dominant events in the data base. This additional check will aid in ensuring consistency in the analysis.

Discussion of Sensitivity to Input Parameters

A sensitivity analysis was conducted to investigate the variability in results to assumed values of input parameters. The sensitivity studies demonstrate the effect of maximum event size, source zone geometry, intensity attenuation, intensity-acceleration relations, and the effect of different I-M_b relations. An important result of the study shows that an intensity-based analysis and a magnitude-based analysis produced essentially the same result for exceedance frequencies for sustained acceleration. The result of the sensitivity study is evidence of the degree of variability in the seismicity curves, as associated with modeling uncertainty.

CHARACTERIZATION OF GROUND ACCELERATION

Effective Acceleration

Comments on the choice of effective acceleration are reserved for the review of Section 7.9.4.

Upperbound for Sustained Acceleration

Arguments are presented to determine an upper-bound estimate on sustained acceleration. The basis of these arguments follows consideration of past experience, and the need to limit the mathematical model that describes the dispersion in ground motion.

In order to define the upper-bound accelerations, the study chooses to use an approach that does not consider the intensity-acceleration model employed in the hazard model. Instead, sustained acceleration data reported by Nuttli is used. Although we agree that other arguments can be applied to define the limit on the extreme value, the basis provided here (Nuttli data) suggests that an acceleration attenuation model would have more appropriately been defined on intensity and calculated sustained acceleration.

As discussed at the PLG meeting, it would have been more appropriate to truncate the lognormal distribution within the hazard analysis. However, it was also agreed, and later verified in a separate calculation that the method of truncation is conservative in that the annual frequencies of exceedance of accelerations below the truncation level will be higher.

CONCLUSIONS

To conclude our review of this section, we judge that the analysis has not adequately represented the uncertainty in the hazard analysis in three general respects; first, all reasonable hypotheses considered explicitly in the analysis were not included in a statement of the uncertainty in the frequency of exceedance curves; second, important alternative hypotheses were not considered in the analysis, specifically consideration was not given to a Ramapo fault zone; and thirdly, we feel that the uncertainty and the mean of the distribution on the upper-bound intensity should be increased.

SECTION 7.9.3 STRUCTURAL MECHANICS ASSOCIATES, INC., FRAGILITY STUDY

Scope of Review

In this section, the developed of the fragility curves for seismic effects is reviewed. In addition to comments on the text, we reviewed the calculations for selected structures and equipment. The results of the calculation check are discussed in the appropriate section below. The references used in the review are listed below.

References

1. Newmark, N. M. and Hall, W. J., "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," NUREG/CR-0098, May 1978.

2. Wesley, D. A., Hashimoto, P. S., and Narver, R. B., "Variability of Dynamic Characteristics of Nuclear Power Plant Structures," Seismic Safety Margins Research Program, Lawrence Livermore National Laboratory, Livermore, California, NUREG/CR-1661, UCRL-15267, July 1980 (prepared by Structural Mechanics Associates).

3. Structural Mechanics Associates, "Engineering Characterization of Ground Motion," Presentation of Task I, USNRC Research Project, San Francisco, California, December 2, 1981.

4. Riddell, R. and N. M. Newmark, "Statistical Analysis of the Response of Nonlinear Systems Subjected to Earthquakes," Department of Civil Engineering Report UILU 79-2016, Urbana, Illinois, August 1979.

5. Wesley, D. A. and Hashimoto, P. S., "Seismic Structural Fragility Investigation for the Zion Nuclear Power Plant," Seismic Safety Margins Research Program, Lawrence Livermore National Laboratory, Livermore, California, NUREG/CR-2320, UCRL-15380, October 1981 (prepared by Structural Mechanics Associates).

6. Benda, B. J., Johnson, J. J. and Lo, T. Y., "Phase I Final Report--Major Structure Response (Project IV)," Seismic Safety Margins Research Program, Lawrence Livermore National Laboratory, Livermore, California, NUREG/CR-2015, Vol. 5, UCRL-53021, Vol. 5, August, 1981.

7. Wesley, D. A. and Hashimoto, P. S., "Nonlinear Structural Response Characteristics of Nuclear Power Plant Shear Wall Structures," Transactions of the 6th International Conference on Structural Mechanics in Reactor Technology, Paris, France.

8. Pickard, Lowe, and Garrick, Inc., "Zion Probabilistic Safety Study," NRC Docket Number 50-295 and 50-304, 1981.

9. ASCE, Structural Analysis and Design of Nuclear Plant Facilities, Manuals and Reports on Engineering Practice, No. 58, 1980.

10. Ang, A. H-S and Newmark, N. M., "A Probabilistic Seismic Safety Assessment of the Diablo Canyon Nuclear Power Plant," Report to the USNRC, November 1977.

11. Kennedy, R. P., et al., "Subsystem Fragility," Seismic Safety Margins Research Program, (Phase 1), Lawrence Livermore National Laboratory, Livermore, California, NUREG/CR-2405, UCRL 15407, February 1982 (Prepared by Structural Mechanics Associates).

SECTION 1. INTRODUCTION

Differences between Units 2 and 3 were noted on IPPSS report page 1-4. In addition, the Unit 2 diesel generator building and portions of the primary auxiliary building are steel frame structures.

The completeness of the components listed in Table 1-1 was checked indirectly by Sandia in their review of the seismic fault trees in IPPSS Section 7.2.

SECTION 2. GENERAL CRITERIA FOR DEVELOPMENT OF MEDIAN SEISMIC SAFETY FACTORS

SECTION 2.1 DEFINITION OF FAILURE

We accept the definition of failure used in this study.

SECTION 2.2 BASIS FOR SAFETY FACTORS DERIVED IN STUDY

We noted on IPPSS report page 2-4 that there was a general lack of detailed information available for this study concerning seismic fragility of structures and equipment. As discussed in Chapter 2, we believe that more detailed analyses should be concluded for structures/equipment which are dominant contributors to the offsite consequences.

SECTION 2.3 FORMULATION USED FOR FRAGILITY CURVES

We believe that the mathematical presentation in this section tends to confuse the casual reader. Because of the inherent simplicity of the method, we offer the following explanation of how it works.

It is assumed in the analysis that the capacity of a structure or equipment, in terms of ground acceleration, is lognormally distributed. Thus, the frequency of failure is a function of three parameters: (1) the median capacity value, $\check{A}$; (2) the logarithmic standard deviation for capacity, β_r, and (3) the ground motion input acceleration value. Note that any randomness in the ground motion value or building or equipment response is included in the β_r value. Figure 7(a) shows the capacity frequency density function which is determined by $\check{A}$ and β_r. If the ground motion value is A_g, then failure occurs for all values of A less than A_g. Thus, the frequency of failure is just the area under the density function between A equal to 0 and A_g. We could stop at this point and just use this procedure to obtain various values of frequency of failure (for different A_g values) and plot the fragility curve as shown in Figure 7(a).

The problem is that $\check{A}$ is not known with certainty. (It is assumed that the logarithmic model and β_r value are known in a relatively certain sense). Thus, a second lognormal distribution for $\check{A}$ is used to quantify the uncertainty for this parameter. It is determined by two parameters: the median value, $\check{\check{A}}$, and the logarithmic standard deviation for uncertainty in the median value, β_u. The probability density function for $\check{A}$ is shown in Figure 7(b).

Now depending on what value of $\check{A}$ is picked from the distribution on $\check{A}$ (see Figure 7(b)), a corresponding fragility curve can be calculated (see Figure 7(a)). For example, if the 95 percent probability fragility curve was desired, then $\check{A}$ would be selected such that there is a 0.95 probability that a larger median value would occur. If $\check{\check{A}}$ is 0.77g and β_u = 0.39, then for the 0.95 probability level $\check{A}$ = .4g. This value comes from the following equation, which is the mathematical representation of the solution shown in Figure 7(b):

$$\check{A} = \check{\check{A}} \exp\left[\beta_u \, \phi^{-1}(1 - p)\right]$$

where $\phi(\cdot)$ = Standard cumulative normal distribution and ϕ^{-1} is the inverse function

p = Probability value (e.g., 0.95)

Now, if the fragility frequency of failure value, assuming β_r is 0.36 is desired corresponding to a ground acceleration A_g equal to 0.4g, the answer can be found from the lognormal distribution with median value to 0.4g (see Figure 7(a) and β_r equal to 0.36). The answer is 0.50 and is found from the following equation:

$$F(A_g) = \Phi\left[\frac{\ell n\ A/\check{A}}{\beta_r}\right]$$

SECTION 3. <u>DIFFERENCES BETWEEN CURRENT METHODS AND CRITERIA USED FOR INDIAN POINT FOR SEISMIC QUALIFICATION OF STRUCTURES AND EQUIPMENT.</u>

SECTION 3.1 <u>EARTHQUAKE LEVEL SPECIFIED FOR DESIGN</u>

No comments are made for the introductory paragraphs.

SECTION 3.2 <u>FREE FIELD STRUCTURAL RESPONSE SPECTRUM ANCHORED TO PEAK GROUND ACCELERATION</u>

It is not obvious why a_s (sustained acceleration or damage-effective) can be used to anchor the ground response spectral shapes from Regulatory Guide (RG) 1.60 which is based on peak response acceleration. It would be more appropriate to redo the statistical analysis using recomputed spectral shapes for the earthquake time histories normalized to the a_s response level (as opposed to peak ground acceleration as done in the original study for RG 1.60).

In the text, two methods for defining design spectra are recognized: specifying site dependent spectra, or using broad-banded spectra such as in Regulatory Guide 1.60. The IPPSS risk analysis used broad-banded spectra. By this selection a source of modeling error is created in the analysis.

In the IPPSS report there is no uncertainty component for variability in the response spectra at all, only randomness. If this were true, then there would be no motivation to ever conduct site studies to develop site-specific spectra. Remember that randomness is irreducible and the IPPSS report broad-banded ground response spectra have no uncertainty. Based on discussions at the meeting with PLG, it was suggested that assuming variability to be evenly divided between randomness and uncertainty would be a reasonable division. The effect on the total parameter values for Indian Point structure/equipment (i.e., median, β_r, and β_u) is not significant.

SECTION 3.3 DAMPING

The damping values given in IPPSS report Table 3-1 are reasonable values for structures and equipment items when the applied stress is near yield. These values are the same as values recommended by Newmark and Hall (Ref. 1). A study of the sensitivity of response of the Zion Auxiliary building for different effects was conducted for the Lawrence Livermore National Laboratory Seismic Safety Margins Research Program (SSMRP) program (Ref. 2). As part of this study, the effect of damping on structure response was investigated. It was found that structure response for a particular earthquake time history (or set of time histories) is weakly affected by damping in the range of 3 to 10 percent. Variation of the median response value was less than 25 percent in this range. For instructure response spectra (which affects equipment response) the damping of the structure had a minor effect except near the fundamental frequency of the structure where the difference was approximately a factor of 2 between the response for 3 and 10 percent damping. This last result indicates that the fragility curves for equipment or substructures with natural frequencies near the fundamental frequency of a supporting structure should reflect the expected structural damping.

From discussion at the meeting with PLG, it was verified that all structures which support safety-related equipment will probably yield before the equipment capacity is reached. This substantiates using yield level damping values for determining structure response.

SECTION 3.4 LOCATION AT WHICH FREE FIELD GROUND RESPONSE SPECTRA ARE SPECIFIED

We agree with the assumption that the free field motion is the same as the input motion at the base slab foundation level for the Indian Point site.

SECTION 3.5 SOIL-STRUCTURE INTERACTION

We agree with this section.

SECTION 3.6 COMBINATION OF RESPONSES FOR EARTHQUAKE DIRECTIONAL COMPONENTS

We agree that the alternate method, consisting of combining 40 percent of the response in two orthogonal directions of motion with 100 percent of the response in the principal direction, is appropriate to use as a median centered method.

SECTION 3.7 SPECIFICATION OF SEISMIC INPUT FOR PIPING AND EQUIPMENT

It is not clear from the description what differences were found between the algebraic summation procedure and the SRSS procedure for combining modal responses. We assume that these differences, if any, were incorporated into the development of the piping fragility parameters.

SECTION 3.8 LOAD COMBINATIONS

The possibility of a severe event which causes a LOCA, followed by an aftershock should be considered. Pressurization of the containment building may fail the reinforcing steel which would weaken the capacity of the building. If this situation occurs, an aftershock could cause additional damage and possibly failure. Although we doubt that this type of occurrence will contribute significantly to the frequency of failure, the possibility should be analyzed and the results documented.

SECTION 3.9 STRESS CRITERIA FOR SEISMIC DESIGN OF CRITICAL STRUCTURES AND CONTAINMENT

Since in the Indian Point reactor building analysis the reinforcing steel was held to the yield value rather than allowing the full ductile capacity to be developed, the Indian Point design criteria appear to be generally more conservative than the USNRC Standard Review Plan criteria.

We noted in the review of the strength parameters for the containment walls that the strength of concrete in shear was considered in developing the fragility curves.

SECTION 3.10 ALLOWABLE STRESS CRITERIA FOR SEISMIC DESIGN OF PIPING AND MECHANICAL EQUIPMENT

Since above-ground piping were not found to be critical components, this section was not reviewed in detail. Thus, no specific comments are made for this section. However, comments concerning piping as a series system are made for Section 5.2.3.1.

SECTION 3.11 SEISMIC CLASS I ELECTRICAL AND INSTRUMENTATION

No specific comments are made for this section. Comments concerning development of fragility values for electrical components and instrumentation are made later in this chapter.

SECTION 4. STRUCTURES

SECTION 4.1 SAFETY FACTORS, LOGARITHMIC STANDARD DEVIATIONS, AND COEFFICIENTS OF VARIATION

No comments are made for the introductory paragraphs.

SECTION 4.1.1 Structure Capacity

No comments are made for the introductory paragraph.

SECTION 4.1.1.1 Concrete Compression Strength

We noted that the overstrength factor was based on data typically reported for other nuclear power plants rather than data from Indian Point test results. It is our understanding from the meeting with PLG that extra uncertainty was included for this consideration, but found to be small.

It is implied in this section that the strength of test cylinders is similar to the strength of in-place concrete. However, it is stated in this section that some decrease in strength is present in in-place concrete. We believe that the variability between test cylinders and in-place concrete is larger than the variability factors for concrete cylinders. Thickness of concrete members and the availability of moisture contribute to actual concrete strength in concrete members. Our estimate is that a logarithmic standard deviation of at least 0.2 would be appropriate.

What is more relevant to the question of capacity is the properties of in-place reinforced concrete strength which includes factors such as construction joints, boundary condition, shrinkage and creep properties, etc., which can be more important than the $\sqrt{f'_c}$ value for concrete material.

SECTION 4.1.1.2 Reinforcing Steel Yield Strength

The values used were compared with similar values given in Appendix A of Reference 3 and were found to be in agreement.

We feel that it is inappropriate to lump No. 3 through No. 11 bars in the same category. No. 3 bars are stronger per unit area than No. 11 bars. However, larger bars comprise the reinforcement generally found in reinforced concrete members in nuclear power plants. This may create a slightly unconservative bias. However, we judge that the effect of this bias is small.

SECTION 4.1.1.3 Shear Strength of Concrete Walls

The basis for Equation 4-3 given in this section of the IPPSS report was reviewed and we agree that this equation is an acceptable prediction of the ultimate strength of shear walls bounded on four sides by concrete members. We feel that the contribution of reinforcement steel given by Equation 4-5 is questionable; however, we did not review the references which led to its derivation. This equation implies that for an aspect ratio (height/width) of 1.0 the vertical steel has no effect on the strength. We find this hard to believe.

SECTION 4.1.1.4 Strength of Shear Walls in Flexure Under In-Plane Forces

We did not review this section in detail. We have no comments.

SECTION 4.1.1.5 Strength of Steel Frame Structures

We concur that the medium yield strength of 44 ksi for dynamic motions is reasonable; however, due to the uncertainty in strength of plates and webs versus flanges, we would have expected the total variability expressed by the logarithmic standard deviation to be larger than 0.11 (i.e., value corresponding to webs). A value of 0.15 would be more reasonable. However, the small difference between 0.11 and 0.15 is not significant to offsite consequences.

SECTION 4.1.2 Structure Ductility

Figure 4-3 in the IPPSS report shows the relationship between the ductility value and the deamplification factor used to increase the median capacity of shear walls for inelastic energy absorption. It should be noted that the results shown in this figure are based on single-degree-of-freedom (SDOF) elastoplastic systems. At a workshop held in December, 1981, sponsored by the USNRC, SMA presented the results of a research project directed to the development of a basis for selecting design response spectra based on free-field motion (Ref. 3). The results of the analytical studies support the deamplification curves given in Figure 4-3. It was found for one example comparison that the difference between Figure 4-3 (IPPSS report) and the methodology developed by SMA when applied to a broad-banded spectrum was less than 10 percent. The study done for the NRC is based on a different approach than taken by Riddell and Newmark (Ref. 4) which is the basis for Figure 4-3 and thus is a good check.

Both the SMA and the Riddell and Newmark studies were based on SDOF models. As noted in Reference 5, considerable uncertainty exists in the application of these techniques to multi-degree-of-freedom (MDOF) systems. No accepted methods currently exist for applying the deamplification factor for SDOF models to MDOF systems. This problem is particularly complex when localized ductilities contribute significantly to the overall strength of a building.

In addition to the variability in the ductility model (it appears that a value of 0.12 was used) an uncertainty measure should also be included for the inaccuracy of using a SDOF model to predict behavior of a MDOF System. A non-linear MDOF analysis of the auxiliary building was conducted for the (SSMRP) (Refs. 6 and 7). As part of this study, five input time histories were applied to the model until a ductility value of four was reached in the weakest element. The ratio of the peak ground acceleration value at failure (defined at a ductility value of four) to the corresponding value at yield was found to range between 1.33 and 1.60 with a median value of 1.43. In comparison, the method used in the IPPSS report to account for inelastic behavior (Figure 4-3) gives a deamplification factor of 0.43 for 10 percent damping. The inverse of this value is 2.35, which is much larger than the more rational median value of 1.43.

This comparison points out the potential differences which can exist between the response of a MDOF structure and the response as predicted by a SDOF model. Our judgment is that there is a large uncertainty which exists and which should be reflected in the fragility parameters. For the dominant structure contributors to offsite consequences (i.e., impact between the Unit

1 and 2 control rooms for Indian Point and the control building shear wall failure), the inelastic energy absorption factors are close to 1.0. Thus very little energy absorption is being relied upon. We feel that this issue does not impact on the final results. However, in general, the uncertainty for the energy absorbing factor is equal to a β-value of 0.1, which we consider to be too small.

SECTION 4.1.3 Structure Response

We accept the methodology described in this section. We note that soil-structure interaction is left out of the list, but is discussed in Section 4.1.3.4.

SECTION 4.1.3.1 Model Response

This category includes the effects of:

- Input ground spectra
- Damping
- Frequency
- Mode shape

We generally agree with the approach used in this section except for the following areas.

As discussed above (see comments for Section 3.2), a larger uncertainty value should be included for the response spectrum input to reflect the potential error between site-specific spectra and the broad-banded site-independent spectra which were used in the analysis.

There is in general a coupling (dependency) between damping and frequency effects. The logarithmic standard deviation values would be different if a combined value were calculated rather than computing the contributions from frequency and damping separately. We judge that this consideration would have a small effect on the IPPSS results.

The logarithmic standard deviation on frequency was estimated to be about 0.2 for all Indian Point structures. This value is different than the value of 0.3 which was used by SMA in the Zion probabilistic safety study (Ref. 8). The results of a study conducted for the SSMRP, where four mathematical models were developed for the same structure, support using the value of 0.3 (Ref. 6). Since the calculations for the original design analyses were not checked, we feel it is more consistent to use 0.3. The small difference between 0.2 and 0.3 is not significant to offsite consequences.

For the effect of mode shape, a logarithmic standard deviation value of 0.10 was used for all Indian Point Class 1 structures. We agree that this is a reasonable value as long as the model has sufficient detail to predict the response of interest. For example, if a flexible floor slab is lumped at a column line in a finite element model, the uncertainty in predicting vertical response at the center of the floor is much larger as compared to results obtained from a model where the floor slab details are included. It was learned at the meeting with PLG that potential flexibility of elements which may not have been modeled (e.g., out of plane response of walls) was considered in the fragility parameter calculations.

SECTION 4.1.3.2 Modal Combination

The values used for this consideration appear to be reasonable based on the data provided in Figure 4-4.

SECTION 4.1.3.3 Combination of Earthquake Components

The 100 percent-40 percent-40 percent method is discussed in Reference 9 where it is stated that it is more conservative than the SRSS method. However, we feel that either of the two methods, can be used to predict median response.

Comments on parameter values for this effect are discussed below as appropriate for specific structures.

SECTION 4.1.3.4 Soil-Structure Interaction Effects

We agree with this section.

SECTION 4.1.3.5 Response Factor Estimate

We noted that in general the median factor of safety for the response factor is between 0.84 and 1.5 for the concrete structures and as high as 2.3 for steel structures. Many of the median factors are close to unity. The primary contribution to be median value comes from the difference between the design response spectra and the median response spectra used in the IPPSS. Most of the conservatism in the design of structures is due to strength and energy absorption.

No discussion is given in the IPPSS report concerning the basis for separating variability into the randomness or uncertainty components. It was confirmed at the meeting with PLG that the variability separation was based almost entirely on subjective judgment. We believe that this fact should be stated in the IPPSS report so that the reader knows the basis.

SECTION 4.2 REACTOR BUILDING

The calculations for the shear capacity of the containment walls for Units 2 and 3 were reviewed. From the calculation sheets, we conclude that the computations follow the general procedure described in the IPPSS sections.

The strength factor was obtained using half of the total containment wall length. This apprears to be reasonable considering that the unit strength is based on formulas for walls with boundary elements; one could argue that the remaining half length of each wall is performing that function. We note however, that the strength provided by the steel was reduced instead of increased due to the 45° inclination of the reinforcement. The overall strength appears to be about 25 percent larger than the reported values. We also estimated the strength using a different assumption (including projection of the walls normal to the loading direction), which yielded strengths more than 50 percent higher than reported. We conclude that the reported median values are conservative; on the other hand, the variations produced by the various assumptions indicate that the uncertainty has been underestimated.

The 10 percent increase in ductility (from 4.0 to 4.5) due to the diagonal reinforcement is adequate, if not conservative. Diagonal reinforcement will delay the development of diagonal cracks. This fact, in addition to the potential reserve of strength mentioned above, offsets doubts concerning the implicit assumption that onset of significant structural damage occurs at a ductility level of 4.0.

The reported median strength factor for Unit 2 is lower than the value for Unit 3; although Unit 2 has about twice the amount of steel (which increases overall strength by about 25 percent). The additional horizontal load on Unit 2 is caused by the earth backfill. There are no details available in the calculations concerning how the additional force was resisted. We made an approximate calculation of the height of backfill corresponding to the total force in the calculations. It is our impression that the IPPSS estimate is conservative and that the Unit 2 strength factor is higher than reported.

We tried to gain additional insight from the calculations as to how the uncertainty, and in some cases, the overall variability was estimated. The calculation sheets tend to confirm that variability was almost entirely based on subjective judgment. As mentioned before, we do not disagree with this method, given the general lack of information. However, we feel that the reported logarithmic standard deviations are generally smaller than they should be.

Calculations for other failure modes of the containment (e.g., failure of the base unit) were not available. The same situation occurs for the auxiliary pump building. However, we do agree that the critical failure mode of the reactor buildings is damage to the containment walls; thus, the review of other calculations is not necessary.

In regards to the substantial reduction of capacity which would occur due to a LOCA we believe that this possibility should be evaluated (See discussion for IPPSS report Section 3.8 above).

For the 1.1g capacity of the Unit 2 containment building, the corresponding vertical acceleration probably would be less than 1g; thus, it is unlikely that it would be thrown into the air.

SECTION 4.3 AUXILIARY BUILDING

The results for individual wall panels appear to be reasonable. The basis for the statement that gross structural failure has a median capacity greater than 3g should be documented.

SECTION 4.4 UNIT 3 CONTROL BUILDING AND DIESEL GENERATOR BUILDING

The calculations for the capacity of the Unit 3 masonry walls at El. 53'-0" and the masonary walls enclosing the battery room, and the shear capacity of the N-S walls of the control building were reviewed.

Masonry Walls

The calculations for the block walls in Unit 3 based on the retrofitted capacities were reviewed. Our general impression is that the fragility values for the retrofitted walls are based on subjective assumptions. The major contributors to the median safety factor and its variability are strength and ductility. The latter is assumed to be 3.0 which appears reasonable. The strength factor is inferred from Brown's Ferry data (not available to us) and a subjective, probably conservative, modification. We have not pursued this component further because the walls are not a key component. One wall is logically parallel to the diesel generator batteries, and it appears that a moderate change in the wall safety factors would have only a minor impact on the overall plant fragility.

N-S Shear Walls of Control Building

The set of calculations that was provided to us consists of two main parts: a dynamic analysis of the control building-diesel generator building complex tied at elevation 32'; and a strength analysis of the governing walls (earliest expected failure), labeled here and in the IPPSS report as "N-S Shear Walls of Control Building."

We did not check the dynamic analysis in detail. Although, our general impression is that it is adequate for the purpose of the IPPSS. The masses and stiffnesses appear to be properly considered, including details such as openings and plan locations of the walls. By comparing the resulting net seismic forces with the size (length) and location of the walls, it appears that the critical wall was properly selected.

The strength analysis of the critical wall appears to be based on reasonable assumptions (e.g., linear strain distribution) and incorporates pertinent details (e.g., openings, flanges of transverse walls). There is some degree of conservatism present due to the assumed length to height ratio for the critical pier and the assumed load demand, but the effect of changing this assumption would be negligible (from the numerical results point of view). The randomness estimate is consistent with values for other shear walls. The basis for the uncertainty estimate is not documented.

For the energy absorption factor, it was assumed that the structure was rigid (11 Hz) which reduced the median factor to almost unity (1.2). The randomness (β_R = 0.03) does not match the value reported in Table 4-10 (β_R = 0.13). The small randomness in the calculations is also due to the rigid structure condition.

The other contributors to the overall safety factor is the spectral shape which is less than unity because of the structure rigidity. We have no specific comments concerning other potential contributors to the safety factor and its variability. The basic assumptions made to account for variability are consistent with those for other components.

SECTION 4.5 UNIT 2 CONTROL BUILDING

The calculations for the capacity of the Unit 2 control room based on impact between the Unit 1 and 2 roof slabs were reviewed. The calculations for the higher capacity value which assumes that the impact problem is eliminated was not checked. Note that the median capacity of the roof impact mode is 0.27g and is the predominant contributor to the offsite consequences for Unit 2.

The main concern for this building is the possibility of impact with the superheater building of Unit 1 which would occur at a low acceleration level well within the elastic limit of the structural system. A rigorous analysis for this problem would involve random vibration theory. The IPPSS calculations indicate that an SRSS combination of the displacements of the two structures was used. This is probably the simplest acceptable method of evaluation and we agree that the results are reasonable. Our only concern is the possible dynamic "beating" effect that has a period of about 2 to 3 seconds (which is less than the duration of the strong motion caused by the closeness of the principal vibration periods of the two structures).

The main contributor to the safety factor is the relatively small displacements predicted by the median response spectrum (which produces smaller responses than those predicted by the original design spectrum).

We agree that the nominal gap between the buildings can be increased for the additional deformation required to fail the connection between the roof and its supports. Local member flexibility is the basis for increasing the gap.

The deflections were linearly scaled from the results of the elastic dynamic analysis. This is reasonable since yielding will not occur at the level associated with impact.

We did not find any explicit reference in the calculations to document the statement that no combination of out-of-phase motions is expected to cause impact below 0.22g. We judge this lower limit to be slightly higher (0.25g). Note that we consider the 0.22g value to be conservative.

We believe that the IPPSS estimate is conservative in the sense it assumes that the control room is out of operation as soon as the roof welds fail. There may be some margin of safety beyond that point, although it is difficult to assess this belief quantitatively.

SECTION 4.6 UNIT 1 SUPERHEATER STACK

The calculations for the capacity of the Unit 1 superheater stack were reviewed and found to be consistent with the procedures followed for other components. It appears that not all the structural data were available to SMA (e.g., the thickness of the steel plates was backfigured from the allowable buckling stress from the original design results; also, the top diameter of the stack was assumed). The original structural model also was not available (e.g., it is not clear if a rotational spring support or a superheater roof response spectrum was used). Due to these uncertainties and the other assumptions made concerning the dynamic behavior of the shortened stack, we are not able to state whether the evaluation is conservative or not. Since basic information was not available, we did not attempt to perform a check analysis. We believe that the strength factor uncertainty (given as β_u equal to 0.15 which is not documented) is too small considering the analysis that was performed. However, it was noted that the total reported variability (β_c = 0.42) is the largest all the critical components (see IPPSS Table 4-14).

We agree with the assumption that the stack could collapse in any direction, provided the superheater building does not have a dominant direction of vibration (which was the assumption explicitly stated in the calculations). However, if there is a dominant direction caused by either the characteristics of the ground motion or the building, the frequency of hitting a specific structure will change (e.g., either increase or decrease). We judge that any reasonable assumptions would not significantly effect the risk of offsite consequences.

SECTION 4.7 UNIT 1 SUPERHEATER BUILDING

We did not check the calculations for this structure; however, we noted that the inelastic energy absorption factor median value is 3.2 (corresponding to a relatively high ductility value of over 7) and an uncertainty β-value of only 0.10. As discussed above for IPPSS report Section 4.1.2, we feel that a much higher uncertainty value should be used. This is even more applicable to structures with high median inelastic energy absorption factors such as the superheater building.

Since this structure is not a dominant contributor, we do not believe that an increase in the uncertainty will have a significant effect on the frequency of offsite consequences.

SECTION 4.8 UNIT 2 TURBINE BUILDING

Comments for the Unit 2 turbine building are the same as given above for the superheater building.

SECTION 4.9 UNIT 2 DIESEL BUILDING

Comments for the Unit 2 superheater bu.lding are also applicable to the Unit 2 diesel building. Because the dominant capacity for Unit 2 is low relative to the capacity for this structure, a more detailed analysis is not warrented. However, because of the importance of this structure, if the Unit 1 and 2 control room impact problem is eliminated, a detailed analysis of this structure should be conducted.

Since no analysis for this structure was available, it is reasonable in light of the low capacity of the control building to assume that the median capacity is similar to the turbine building; however, the uncertainty should be larger to reflect the lack of structure-specific information.

SECTION 4.10 BURIED CONCRETE STRUCTURES

We concur that the strengths of the buried concrete structures are relatively high.

SECTION 4.11 FUEL STORAGE BUILDINGS

The calculations for this structure were not reviewed.

SECTION 4.12 NONCRITICAL STRUCTURES

r .d on our tour of the Indian Point site, we did not observe any other major structures which could fail and fall on safety-related systems and components.

SECTION 5. EQUIPMENT FRAGILITY

SECTION 5.1 GENERAL APPROACH AND INFORMATION SOURCES

We noted that no new analyses were conducted for equipment items.

SECTION 5.1.1 Information Sources for Equipment

No comments are made for this subsection.

SECTION 5.1.2 Equipment Categories

No comments are made for this subsection.

SECTION 5.1.3 Response Factor Categories

We agree with the categories in this subsection.

SECTION 5.1.4 Structural Response

As noted for Subsection 4.1.3.1, we raised the issue of mode shape ordinate error due to flexibility of a local element or substructure. This is particularly appropriate for development of fragility data for subsystems which are supported by the structure.

Modal combination is not included in the list of variables. This is because the floor response spectra used to design the equipment were developed using a direct integration procedure.

SECTION 5.2 EQUIPMENT CAPACITY FACTORS

Specific comments are made for each of the sections in the following text. In order to assist in determining the implication of issues and questions which are raised, the components listed in Table 5-3 of the IPPSS report were associated with the various report sections. Table 2 lists the IPPSS report sections, components, and median ground acceleration values. Particular attention was given to key equipment (see IPPSS report Tables 7.2-3 and 7.2-7).

In reviewing many of the fragility parameters, it was not clear what specifically constituted the underlying bases. We raise this issue for specific parameters in order to determine which ones are based on data, engineering judgment, or a combination of these sources. For example, one parameter which is common to almost all components is material yield strength. The basis for assuming that the median yield value is 1.25 times the code specified value should be documented (however, this does appear to be a reasonable value). The basis for the variability β_C value of 0.14 and the associated randomness and uncertainty components of β_r equal to 0.1 and β_u equal to 0.1 also should be documented.

It was learned at the meeting with PLG that the separation of variability into its randomness and uncertainty components was primarily based on judgment. We believe that this should be documented in the IPPSS report. In instances where analysis or data form the basis for selecting parameter values, this should be documented. We do not object to determining parameter values subjectively, but feel it is imperative that the reader know what was done.

SECTION 5.2.1 Plant Specific Structural Capacities Derived from Design Reports

It is stated in this section that the logarithmic standard deviations for capacity (i.e., randomness and uncertainty), were derived in the same manner as for structures. We noted for Subsection 4.1.3.5 that the basis for separating the total variability into randomness and uncertainty components for structures is not provided. Our understanding is that this was done primarily using engineering judgment. This should be documented in the IPPSS report.

The ductility factor used for equipment (Equation 5-5) is different from the approach used for structures, which was based on deamplification factors for elastic-perfectly plastic systems (see Figure 4-3 in IPPSS report). For structures, the ductility factor is a function of ductility and damping, while the factor for equipment is a function of only ductility. However, the differences between the two approaches is small.

Since both factors (i.e., for structures and equipment) are for single-degree-of-freedom elastic-perfectly plastic systems, there is inherent error in using these models for multidegree-of-freedom equipment (see comments for Subsection 4.1.2 for the same problem for structures). A logarithmic standard deviation value of 0.2 was used for uncertainty. We suspect the true value is higher and that an additional small value for randomness should also be included.

SECTION 5.2.1.1 <u>Reactor Pressure Vessel</u>

It is not clear from the description exactly how the median strength and variability was calculated. In particular, variability is not documented for the shape factor. The basis used to determine the two logarithmic standard deviation bounds and the basis for the ultimate strength (i.e., versus the yield strength) used in determining the upper-bound strength should be documented.

The variability of Equation 5-5 due to variability of only ductility gives β equal to 0.23 based on the combined value being 0.30 and the variability in the equation itself being 0.20. The value of β equal to 0.23 apparently comes from the following calculation:

$$\beta = \frac{1}{2} \ln \left[\frac{\sqrt{2(3) - 1}}{\sqrt{2(1.5) - 1}} \right]$$

Another way to compute the value is to use a Taylor series expansion approach which gives a median value of F_{μ} equal to 2.27 (compared to 2.24) and β equal to 0.21 (compared to 0.23). Thus, the method used in the IPPSS report gives acceptable values.

SECTION 5.2.1.2 Reactor Pressure Vessel Intervals

The basis for the median shape factor value for collapse moment equal to 1.86 (we noted that $4/\pi$ x 1.144 equals 1.46, not 1.86) should be documented. Also, the basis for assuming that $4/\pi$ is minus 2 logarithmic standard deviation values below the median should be given.

The derivation of the strength factor is not clear from the text. At the start of the section, seismic stresses, for a Housner spectrum anchored to 0.25g for the OBE, were found to be 51.2 percent of the code allowable value of 1.5 S_m. For the DBE, considered to be twice the OBE, the stresses were found to be 1.12 times median yield strength (does this imply that median yield is 1.37 S_m?). The strength factor is then computed to be 1.86 divided by 1.12 or equal to 1.66. It is not clear what ground acceleration value this calculation related to (i.e., 0.15g or 2 x 0.25g). We feel that these issues will not ultimately impact on the user of offsite consequences.

SECTION 5.2.1.3 Steam Generator

The approach used for this component appears to be reasonable. Any small changes in the parameter values will not affect the frequency of core melt analysis since the median capacity is relatively high.

SECTION 5.2.1.4 Reactor Coolant Pump

The capacity for this component is relatively high.

SECTION 5.2.1.5 Pressurizer

The calculations for this component were reviewed. The calculations contain the analysis for the equipment capacity factor. There was no information about the derivation of the response factor. The response factor computations are discussed for IPPSS Section 5.3.2.1.

In determining the capacity factor, two conditions were analyzed: the capacity of the base flange and the capacity of the bolts. We did not checked numerically the computations, but we agree with the general flow of calculations and the details that were considered. The sources of information for the structure data are referenced except for the DBE load which is only stated. We believe that sufficient detail was considered to produce reliable results.

Although we do not have complete information we generally agree with the the median acceleration capacity. The variability parameters appear to be consistent with other IPPSS results.

SECTION 5.2.1.6 Control Rod Drive Mechanisms

We estimate the median ground acceleration value for this component to be between 2g and 3g, which agrees with the IPPSS report.

SECTION 5.2.1.7 Reactor Coolent Piping

Basically the capacity of this component is the same as for the reactor pressure vessel except thermal stresses have been removed since they are considered to be self-limiting. Even if the thermal stresses were included, the capacity of the component is very high and thus, will not affect the frequency of offsite consequence calculations.

SECTION 5.2.1.8 Safety Injection Pump

In developing the median strength factor value of 1.64, a shape factor of 1.5 and a yield strength factor of 1.25 are assumed. (i.e., 1.64 = (35 x 1.50 x 1.25/40)). The shape factor value should be documented in the report.

We understand, based on the meeting with PLG, that the variability logarithmic standard deviation for material equal to 0.14 is based on data. This fact should be documented along with the data or literature source where the analysis of the data can be found.

In developing the uncertainty for the strength factor, uncertainty also should be included for the fact that the pump material is not specified and an assumption that it is carbon steel was made. Although the shaft/bearing interaction median capacity is slightly larger, variability for this failure mode should be computed. A large variability for a slightly weaker mode may produce a larger probability of frequency of failure at acceleration values below the median.

SECTION 5.2.1.9 Residual Heat Exchanger

In developing the uncertainty for the strength factor, uncertainty also should be included for the fact that the heat exchanger shell material is not known and an assumption that it is 516-Gr 60 was made.

In an identical PRA analysis for the Zion plant, the possibility of bucking in the shell was considered. In this case, no inelastic energy absorption was assumed (Ref. 8). In the IPPSS, a median energy absorption factor of 1.73 was used corresponding to anticipated ductile behavior. Since the heat exchangers are essentially the same in both plants, only one of the assumptions should be correct.

Since the median capacity for this component is relatively high, the resolution of these issues will not affect the frequency of offsite consequences.

SECTION 5.2.1.10 Component Cooling Heat Exchanger

The capacity for this component is relatively high.

SECTION 5.2.1.11 Accumulator Tanks

The capacity for this component is relatively high.

SECTION 5.2.1.12 Boron Injection Tank

The capacity of this component is relatively high.

SECTION 5.2.2 PLANT-SPECIFIC FUNCTIONAL CAPACITIES DERIVED FROM DESIGN REPORTS

We believe that eliminating inelastic energy absorption is conservative; however, it may be more appropriate in some cases to include the effect of ductility. In these cases, the median capacity would be higher, which would be offset to some degree by a higher uncertainty value to reflect the inability to determine when a functional failure occurs.

SECTION 5.2.2.1 Containment Fan Coolers

Based on our meeting with PLG, we learned that the worst case manufacturing tolerance stack-up would occur approximately 2 in 1,000 cases (i.e., approximately -3σ) based on manufacturing experience. We feel that this should be documented in the IPPSS report along with the data or literature source for the data. The basis for other assumptions in this section should also be documented.

Calculation for the containment fan coolers were reviewed. The calculations show the development of the safety factors and associated logarithmic standard deviations. The development follows the procedure given in the IPPSS report, and the variabilities are consistent with the general assumptions used throughout the IPPSS report. The selection of the critical strength factor from three possible failure modes is documented; the main data, however, are only referenced and not otherwise given. From this information we are unable to conclude about the accuracy of the strength factors. All we can state is that a systematic procedure was used.

We note, however, that this equipment is logically in parallel with two other component paths. The impact of changes to the capacity of the fan coolers will be neglegible to the overall plant fragility.

SECTION 5.2.2.2 Residual Heat Removal Pumps (RHR)

The basis for the assumptions made in this section should be documented.

SECTION 5.2.3 GENERIC STRUCTURAL CAPACITIES DERIVED FROM DESIGN CRITERIA

No comments are made for the introductory section.

SECTION 5.2.3.1 Piping and Supports

We believe that it can be unconservative to base the fragility of the piping system on the single component type most likely to fail. This procedure implicitly assumes that the individual components are perfectly correlated. In reality, a piping system consists of a series of components whose capacities and responses are each partially dependent (Ref. 10). One approach for including this effect would be to determine an equivalent number of independent components, which would be based on the type of elements (e.g., butt welds, their number, location, etc.). Because piping systems can be very long, it is prudent to make a best estimate of the effect of dependency even if it is only based on engineering judgment.

In discussions with PLG it was stated that most piping systems have only one or two critical components. The rest of the components are generally overstressed. If this is the case, then it does not matter whether or not partial independence is assumed. We believe that it is prudent to look at each safety-related piping system to determine that it is in fact reasonable to assume that only one component controls the capacity.

It is not clear in later development of the fragility parameters if the effect of the combination "0.75i" in the stress acceptance equation was incorporated.

SECTION 5.2.3.1.1 Support Failure Modes

The decision to base the fragility analysis on supports that only carry seismic load implicitly assumes that the total applied stress as a percentage of the design stress is essentially the same whether normal stresses are present or not. This assumption appears to be reasonable.

SECTION 5.2.3.1.2 Piping Fragility

In developing the ratio of static collapse load to allowable design load (i.e., P_L/P_D), if we assume a ratio of S_h to yield to be between 0.625 and 0.9 (along with the other factors given in this section), we find that P_L/P_D ranges between 1.62 and 2.33. If we then incorporate the various P_N/P_D and P_{OBE}/P_D ratios given in this section into equation 5.4, we obtain a median value of 4.6 (compared to 5.9) and a β_S of 0.40 (compared to 0.27). We believe that these differences would not affect the frequency of offsite consequences.

SECTION 5.2.3.1.3 Support Fragility Description

It was learned at the meeting with PLG that the logarithmic standard deviation value of 0.42 for the strength factor was obtained by establishing a lower bound factor of safety using a minimum strength (code yield stress of 25 ksi reduced 15 percent for welding or threads, i.e., 21.2 ksi) and a maximum load stress of 1.1 times design stress which is 50/4 x 1.2 x .75 = 11.25 ksi where 1.2 is a short term load factor and 0.75 is <u>also</u> a factor for threaded connections. The lower bound factor of safety is then equal to 21.2/(1.1 x 11.25) or 1.7. Then β is equal to 1/3 (ln 5.9/1.7) or 0.42, where 5.9 is the median factor.

We believe that this is incorrect since the effect of threaded connections appears to be included twice and the code yield stress is not increased by a factor of 1.25 to a median value. A more rational β-value would be 0.28 instead of 0.42. On the other hand, a 3σ range seems high. If a more defendable 2σ range is used, the β-value is back to 0.43. Thus, we concur with the value used.

SECTION 5.2.3.1.4 Governing Criterion for Piping

Except for the issue of dependence between piping system components, we feel that the issues raised will not affect the frequency of core melt analysis. However, as stated above, since the piping systems can be long with many components (hence potentially many locations for failure), the effect of dependency could lower the effective piping capacity sufficiently such that piping becomes an important component. We are willing to accept the argument, in general, that only one or two components are stressed to allowable values in a piping system; however, we feel that each critical piping system should be reviewed to determine that this assumption is appropriate.

SECTION 5.2.3.2 Generic Fragility for Other Equipment That Fails in a Structural Mode

It appears that the combined normal plus OBE load could range as high as 1.3 times (not 1.1) the allowable design load.

We reviewed the calculations for the median strength factor and the associated logarithmic standard derivations. We do not agree entirely with the method used, but feel that the values obtained are reasonable.

We note in IPPSS report Table 5-3 that median ground acceleration values for low capacity components in this category are as follows:

	Unit 2	Unit 3
Condensate Storage Tank	1.28 g	1.28 g
RWST	0.70 g	0.70 g
Diesel Generator Oil Storage Tanks	1.14 g	1.14 g
Batteries and Racks	1.37 g	1.07 g
Service Water Pumps	2.47 g	2.47 g
Spray Additive Tank	1.01 g	1.01 g
Duckwork and Dampers	1.12 g	1.12 g

We reviewed the calculations for the condensate storage tank, the RWST, the diesel generator oil storage tanks, the batteries and racks, and the service water pumps.

Diesel Storage Tanks

According to the documentation provided to us by SMA, the tanks were not analyzed. The tanks were assigned a generic capacity for heavy equipment (page 5-43, 5-44 of IPPSS which is 1.14g). For an underground structure this capacity is credible and probably conservative, but somewhat arbitrary.

The most important component for Unit 3 is the diesel generator fuel oil tanks which is the dominant contributor. We believe that specific fragility calculations should be performed for this component.

Other Components

We reviewed the calculations for the battery racks, service water pumps, RWST, and the condensate storage tanks. The method of development of the safety factors is consistent with the IPPSS report. Actual strength calculations were not found but the calculations point out the sources of information (i.e., previous Zion plant PRA results) or state previously calculated strengths, presumably from separate computations (e.g., condensate storage tank). The randomness and uncertainty measures appear to be consistent with others used in the report; although this is difficult to check on an item-by-item basis. While the accuracy of individual values are probably low, it is our opinion that in combination they represent a systematic way of assembling the basic information.

We note that, except for the battery racks in Unit 2, all the components discussed in the previous paragraph have at least one degree of redundancy, according to the fault trees.

SECTION 5.2.4 Capacities Derives from Tests for Higher Seismic Zone Criteria

The capacities for components that are included in this category are relatively high such that any small changes in the parameter values will not affect the frequency of offsite consequence analysis.

SECTION 5.2.5 Generic Capacities Derived From Military Shock Test Data

No comments are made for the introduction.

SECTION 5.2.5.1 Electro-mechanical Equipment

It is not clear from Table 5-3 which component fragility values were developed based on Army Corps of Engineers test data for electrical-mechanical equipment. This should be documented in the IPPSS report.

Comments concerning capacities determined using data from the SAFEGUARDS program tests is discussed in the next subsection.

SECTION 5.2.5.2 Electrical and Control Equipment

Reference 11, which was prepared for the SSMRP by SMA, gives background on reduction of data from the SAFEGUARDS program. This does not represent an independent check since both this IPPSS report section and Reference 11 were prepared by the same authors. We generally concur with the development of hazard curves for relay chatter and breaker trip. However, we are uncomfortable with the general conclusion that failure occurs at a level three times the fragility level for recoverable interruptions.

Our position is based on two points, First, the duration of the input in the SAFEGUARDS tests was only 2 seconds long. During a large seismic event, the duration of motion will be on the order of ten to twenty seconds long. We can conceive of failure at a lower acceleration level due to the effects of duration. Second, we are concerned whether the equipment tested in the SAFEGUARDS program is representative of the specific safety-related equipment at Indian Point.

We agree that nonrecoverable failure is higher than relay chatter or breaker trip. However, we question whether the strength is a factor of three higher, or possibly only fifty percent higher in some specific cases. We recommend that if a particular electrical or control component is a dominant contributor (or potentially a contributor) to offsite consequences, that a specific analysis be performed for that piece of equipment. At a minimum, the particular components (i.e., switches and breakers) should be compared to the units tested in the SAFEGUARDS program. If the units are different, then an independent basis for determining the fragility should be found.

For the case of Indian Point, the following equipment from Tables 5-3 and 5-4 are potential contributors. The values given are median ground acceleration capacity values for recoverable interruption.

Equipment	Unit 2 Symbol	Unit 2 Capacity	Unit 3 Symbol	Unit 3 Capacity
Diesel Generator Controls	18	1.30g	20	1.30g
120 VAC Distribution Panels	30	1.65	17	1.19
480 VAC Motor Control Centers	31	1.65	16	1.17
480 VAC Switchgear and Station Transformer	-	-	29	1.51

For Unit 2, any reasonable failure capacity is much larger than the dominant contributor which has a median capcity at 0.27g; thus, we see no problem for Unit 2. However, for Unit 3 the equipment associated with the diesel generator has recoverable capacities similar to the fuel oil tanks which is a dominant contributor. We recommend that the capacities of three times the values for equipment listed above for Unit 3 be confirmed.

SECTION 5.2.6 Generic Capacities for Valves

Based on discussion at the PLG meeting and inspection of Indian Point, we concur that the capacities for the safety-related valves at Indian Point are relatively high.

SECTION 5.2.7 Cable Trays

The calculations for the cable trays at Unit 2 were obtained. These refer back to this section of the IPPSS report. However, we feel that the capacities of individual cable trays (and supports) are reasonable.

As discussed above, we feel that the capacity of a single support may be unconservative since there are many cable tray supports in series which are safety-related. Since they are not perfectly dependent, the frequency of failure may be less than for a single support. Based on our inspection of Indian Point and our review, we believe that potential failure of cable trays are not dominant contributors to offsite consequences.

SECTION 5.2.8 Offsite Power

We agree that the median capacity of ceramic insulators is low and it is reasonable to assume in the systems analysis that they have failed.

SECTION 5.2.9 Diesel and Gas Turbine Generators

Based on our inspection of Indian Point, we agree that the fragility of these units are dominated by the control panel fragility. See comments for Section 5.2.5.2 above.

SECTION 5.3 Equipment Response Factors

We have no comment on the introduction to this subsection.

SECTION 5.3.1 Plant-Specific Equipment Qualified by Dynamic Analysis

The residual heat exchanger was selected as an example to demonstrate the methodology of deriving the response factors. For clarification and use in the review of later sections, the enveloped floor response spectrum used in the design, and the applicable IPPSS floor response spectra should be provided. No discussion was given as to what method was used to determine the applicable IPPSS floor spectra. Depending on the method used to develop the applicable Indian Point floor response spectra, the basis for determining uncertainty due to modeling may be different.

SECTION 5.3.1.1 Spectral Shape

The basis for the variability value of 0.10 should be documented.

SECTION 5.3.1.2 Qualification Method

We agree that the response spectrum method is median centered with variability equal to zero.

SECTION 5.3.1.3 Damping

The approach used in this section is reasonable. Since the response spectra are not available, we did not review the calculations.

SECTION 5.3.1.4 Frequency

The approach used in this section is reasonable. Since the response spectrum was not provided, we did not review the calculations.

SECTION 5.3.1.5 Mode Shape

We agree that the response factor for mode shape is 1.0. The assumption that the logarithmic standard deviation is 0.15 for multi-degree-of-freedom and 0.1 for single-degree-of-freedom systems is not substantiated in the text or in the referenced report (Ref. 50). Clarification of these values is needed. It is unrealistic to assume that the variability is constant for all equipment.

It has been assumed here and in previous sections that the residual heat exchanger responds predominantly in a single mode. No basis is provided to support this. However, we anticipate that any change to the mode shape parameter will have a small effect on the frequency of offsite consequences.

SECTION 5.3.1.6 Mode Combination

The approach used in this section is reasonable.

SECTION 5.3.1.7 Combination of Earthquake Components

It appears that the variability value of 0.09 comes from assuming a 3 difference between the median value of 1.08 and the maximum value of 1.41. We agree with the median response factor value that was derived.

SECTION 5.3.1.8 Combined Response Factor and Variability

We have no additional comments for this subsection.

SECTION 5.3.2 Plant Specific Equipment Qualified by Dynamic Analysis

We have no additional comments for this section.

SECTION 5.3.2.1 Flexible Equipment

We reviewed the calculations for the pressurizer as discussed below.

SECTION 5.3.2.1.1 Qualification Method
SECTION 5.3.2.1.2 Damping

It was stated but not referenced in the calculations, that the DBE static load was 0.96g. This, and a 5 percent damping spectrum yielded a factor of 2.4 which is consistent with the combined factors in the IPPSS report (1.28 for 1 percent damping and 1.88 to reduce it to a 5 percent damping value). The damping variabilities in the calculations seem to be an earlier version of those used in the final report; differences are minor. Provided the input data are correct, we agree with the results.

SECTION 5.3.2.1.3 Frequency

SECTION 5.3.2.1.4 Mode Shape

SECTION 5.3.2.1.5 Mode Combination

Frequency, mode shape, and mode combination are treated in the calculations as a general modelling error. Again the calculations seem to be an earlier version of the final reported values in the IPPSS report. We agree with the values in the final report.

SECTION 5.3.2.1.6 Combination of Earthquake Components

Since the EW floor response is small compared to the perpendicular direction (0.4g vs 0.19g) the 100 percent-40 percent-40 percent method gives a median value barely above the 0.4g value for the strong direction. The vector sum was taken as a worst case and the 0.4g value as the best case; then the median was obtained by a logarithmic average using a 2 standard deviation range. The value obtained is conservative. The effect of this result on the fragility of the pressurizer is about 4 percent.

SECTION 5.3.2.1.7 Combined Response Factor and Variability

We have no additional comments for this section.

SECTION 5.3.2.2 Rigid Equipment

We agree that the only response factors to be considered for rigid equipment are the qualification method and earthquake component.

SECTION 5.3.2.2.1 Qualification Method

The applicable floor response spectrum was not available to verify the qualification method factor of 5.56.

We agreed that there is a small variability associated with the qualification method factor, with the exception that there may be an uncertainty component due to the method of determining the floor response spectrum. This concern was also raised earlier in comments for Section 5.3.1.

SECTION 5.3.2.2.2 Earthquake Component Combination

We reviewed the calculations for this factor. Although we do not entirely agree with the method used, we feel that the median and variability values are reasonable.

SECTION 5.3.2.2.3 Combined Factors and Variability

We have no additional comments for this section.

SECTION 5.3.3 Plant-specific Equipment Qualified by Test

We agree that the response factors cited are those which should be considered for equipment qualified by testing for the IPPSS.

SECTION 5.3.3.1 Spectral Shape

We have no additional comments for this section.

SECTION 5.3.3.2 Boundary Conditions

We agree that the test conditions can be assumed to be median centered with respect to the conditions at the plant. We note, however, that different failure mechanisms may exist for the supports in the IPPSS plant. For example, in the tests, bolt support failure was a possibility while under plant conditions; this is reportedly not a likely event. However, the report (Ref. 50) does not provide variability for this difference. During our tour of the Indian Point facilities we did see numerous panels which were bolted to the floor slab. Hence, we feel that the test conditions may be very similar to the Indian Point construction.

SECTION 5.3.3.2 Damping

We agree that the median response factor due to damping is 1.0; however, insufficient information was provided to verify the derivation of the variability factors.

SECTION 5.3.3.3 Frequency

Insufficient information was provided to verify the derivation of the response factor and variability. The basis for assuming that the response corresponding to the frequency range 5 to 10 Hz is a $\pm 2\sigma$ range should be documented. This assumption results in a low logarithmic standard deviation on response.

SECTION 5.3.3.4 Multi-mode Effects

The basis for assuming the range 1 to 1.5 to be $\pm 2\sigma$ above the median should be documented.

SECTION 5.3.3.5 Earthquake Component Combination

We reviewed the calculations for this factor. Although we do not entirely agree with the method used, we feel that the median and variability values are reasonable.

SECTION 5.3.3.6 Combined Response Factors and Variability

We have no additional comments for this section.

SECTION 5.3.4 Response Factors for Generic Categories of Equipment

The basis for defining various types of equipment as generic, particularly in situations where the systems are complex, should be provided.

SECTION 5.3.4.1 Piping 6" in Diameter and Less

The approach used to establish the median factor and variability due to qualification method (note that the section number 5.3.4.1.1 evidently was dropped) is reasonable.

SECTION 5.3.4.1.2 Damping

A simple assumption was used to determine the frequency of all piping systems. Although the estimate appears to be reasonable, there is an additional uncertainty component in the method used to develop the response factor and variability, particularly since the factor is being applied to all piping situations. It is anticipated that only small changes would result if additional uncertainty was added for this effect.

SECTION 5.3.4.1.3 Frequency, Mode Shape and Mode Combination

We agree that these variables are all contained in the qualification method and its variability.

SECTION 5.3.4.1.4 Combinations of Earthquake Components

The method for developing the parameter values for this section is the same as for Section 5.3.2.2.2, which we reviewed and concur that the parameter values are reasonable.

SECTION 5.3.4.1.5 Total Response Factor and Variability

We have no comments for this section.

SECTION 5.3.4.2 Piping 6" in Diameter and Greater

We agree that the factors cited are those to be addressed for this class of piping

SECTION 5.3.4.2.1 Spectral Shape

We have no additional comments for this section.

SECTION 5.3.4.2.2 Qualification Method

We have no comments for this section.

SECTION 5.3.4.2.3 Damping

The basis for choosing 10 Hz as the frequency to develop the response factor for damping should be documented. Given the various piping configurations, a single frequency is not appropriate. In addition to the variability associated with the randomness due to material effects, there would also be a component of uncertainty due to the method for selecting pipe frequencies and the variability in frequencies throughout the plant.

SECTION 5.3.4.2.4 Frequency

We agree that the modal analysis is median centered. The basis for using 10 Hz as the median value should be documented.

SECTION 5.3.4.2.5 Mode Shape

We agree that the response spectrum analysis is median centered. The basis for the logarithmic standard deviation of 0.15 should be documented. The same comments we made for mode shape for structures (see Section 4.1.3.1) also apply here.

SECTION 5.3.4.2.6 Mode Combination

We have no additional comments for this section.

SECTION 5.3.4.2.7 Combination of Earthquake Components

We have no additional comments for this section.

SECTION 5.3.4.3 Valves

We agree that valves can be considered rigid for frequencies above 20 Hz. No reference is provided, however, to support the assumption that all valves have frequencies greater than 20 Hz. We agree that the response acceleration of a rigid valve will be equal to the acceleration of the pipe at the point of attachment. We feel that a similar set of parameters could be developed for valves similar to the developement for piping less than 6 inches in diameter.

SECTION 5.3.4.3.1 Qualification Method

The basis for using a range equal to the ZPA to 1.5 times the peak spectral acceleration as a $\pm 2\sigma$ range should be documented.

SECTION 5.3.4.3.2 Damping

We agree that the factor is median centered; however, there should also be a component of variability attributable to the valve, in addition to that associated with the piping, albeit this may be small.

SECTION 5.3.4.3.3 Frequency, Mode Shape and Mode Combination

We have no additional comments for this section.

SECTION 5.3.4.3.4 Combination of Earthquake Components

We agree that this factor is identical to that determined for piping. We have no additional comments for this section.

SECTION 5.3.4.4 Floor and Wall-Mounted Equipment With Generic Capacities

We have no additional comments for this section.

SECTION 5.3.4.5 Cable Trays

We agree with the method for determining the response factor.

SECTION 5.4 Structural Response Factors

Comments concerning these factors are made for the Sections 4.1.2.1 through 4.1.2.6 from Chapter 4 of IPPSS Section 7.9.3.

SECTION 5.5 Fragility Description

No comment.

SECTION 7.9.4 STRUCTURAL MECHANICS ASSOCIATES, INC. DAMAGE-EFFECTIVE GROUND ACCELERATION

Scope of Review

The basis for converting peak ground acceleration to damage-effective ground acceleration and the upper-bound cutoff on effective acceleration are reviewed. Two additional sources were also read and used in the review of these two concepts.

References

1. Kennedy, R. P., "Peak Acceleration as a Measure of Damage," Presented at Sixth International Seminar on Extreme-Load Design of Nuclear Power Facilities, Paris, France, August 1981.

2. Kennedy, R. P., Tong, W. H. and Short, S. A., "Earthquake Design Ground Acceleration Versus Instrumental Peak Ground Acceleration," SMA 1205.01R, Structural Mechanics Associates, Newport Beach, California, December 1980.

SECTION 1. INTRODUCTION

We concur with the concept that near-field lower magnitude earthquakes are generally less damaging than far-field magnitude events with the same instrumental peak ground acceleration value. We raise several issues, which are discussed in the next section, which question how this concept was applied in the IPPSS.

SECTION 2. EFFECTIVE PEAK VERSUS INSTRUMENTAL PEAK AND SUSTAINED PEAK ACCELERATIONS

As part of our review for this section, we read Reference 1, which explained in more detail the concepts discussed in Section 7.9.4. Reference 1 in turn refers to a report which documents the basis, that for the purpose of predicting elastic response of structure in the 2 to 10 Hz frequency range, median broad-banded amplification spectra (such as used in developing the fragility curves) are more accurately anchored to an acceleration value equal to 1.25 x A_{3F} (Ref. 2). In Reference 2, twelve earthquake response spectra are compared to the mean plus one standard deviation WASH-1255 amplification spectrum anchored to 1.25 x A_{3F} for each time history.

Visually, the comparison between the two types of spectra (actual and broad-banded) in Reference 2 are convincing. In the 2 to 10 Hz frequency region, the comparison appears to be median centered. However, it is difficult to visually determine what the difference would be if the median amplification spectrum (which was used in the IPPSS report) had been used instead. It would be more comforting if a statistical analysis had been performed to verify that 1.25 is the appropriate factor.

The adjustment of the anchor acceleration value must be done with caution. Near-field low magnitude response spectra tend to be peaked at one (or more) natural frequencies for a particular site. In general, the broad-banded spectrum will be conservative except near the peak of the site-specific spectrum, where it may be just right. Thus, the correction factor F is appropriate in a median sense; however, there is uncertainty which exists for any specific structure. It makes a difference whether a fundamental building frequency is higher or lower than the frequency corresponding to the peak of a site-specific spectrum, in regards to whether significantly less damage will occur for a near-field low magnitude event.

A rational procedure for determining a value for F for a specific structure would be to determine the relative damageability between the best estimate of the site-specific response spectrum and the broad-banded spectrum used in the IPPSS analysis at the fundamental frequency of the structures being considered.

We are also concerned about applying this concept to equipment located in a building without first confirming that it is appropriate to do so. A structure acts as a filter which smooths the incoming seismic time history to produce a more sinusoidal appearing time trace at equipment support locations. Whether the same argument for the factor F can be made for equipment housed in a structure as for structures supported on the ground needs to be documented.

The value of F recommended in this section is equal to 1.25. We believe that even if the value were 1.0 that only a small effect would occur to the frequency of core melt analysis for Unit 2 and a moderate effect for Unit 3. In general we believe that a value of F equal to 1.25 is on the conservative side for structures. For equipment located in structures, which have a capacity below the capacity of the equipment, this value of F is probably also conservative. The argument given by SMA at the meeting with PLG is that the softening of the structure stiffness at high levels of ground motion will decrease the input to the equipment. All safety-related equipment which affects potential offsite consequences falls into this category. This value may not be conservative for certain equipment located on the ground or attached to the base of structures. Equipment, which does not have inelastic energy-absorption capacity or which depends on function capacity, respond more closely to the peak ground acceleration capacity. One example of this type of equipment is the service water pumps which depend on binding of the pump shaft for capacity and which are located at the ground level. However, the capacity of this component is relatively high and eliminating the 1.25 acceleration factor would not significantly change the results of the analysis.

SECTION 3. <u>UPPER BOUND CUT OFF ON EFFECTIVE PEAK ACCELERATION</u>

We agree that the upper-bound acceleration values given in this section are reasonable. It should be noted that these values are <u>conditional</u> specific values of intensity. This section does not attempt to establish upper-bound values on intensity for earthquakes in the region surrounding Indian Point (the argument for a maximum intensity is in IPPSS Sections 7.2 and 7.9.2).

SECTION 7.9.5 RESEARCH TRIANGLE INSTITUTE REPORT WINDSPEED RISK ANALYSIS OF THE INDIAN POINT NUCLEAR GENERATING STATION

Scope of Review

This section of the IPPSS report gives the basis for the tornado and hurricane (including extratropical winds and thunderstorms) wind speed hazard curves. We performed an approximate analysis for tornado effects, which convinced us that the hazard curves are conservative. Dr. Larry R. Russell reviewed in depth the material in the section on hurricanes (see Appendix C). We also offer comments concerning the development of the hurricane hazard curves. Comments concerning extratropical wind hazard curves and the approach used to develop the probabilistic family are also given.

REFERENCES

1. Fujita, T. T., "Workbook of Tornados and High Winds for Engineering Applications," Department of the Geophysical Sciences, The University of Chicago, SMRP Research Paper 165, September 1978.

2. Thom, H. C. S., "New Distribution of Extreme Winds in the United States," J. of Structural Division ASCE, Paper 6038, 1068.

3. Changery Report (Add in Final Report).

SECTION I. INTRODUCTION

We interpret the statement: "no localized wind regime mechanism is assumed to be present," to mean that the effects of topography in the vicinity of the site and the arrangement of buildings at Indian Point were not specifically included in the mathematical models used to develop the wind speed hazard curves given in the IPPSS.

SECTION II. PROBLEM DESCRIPTION

We note below that probabilities were subjectively assigned to the lower and upper bounds. The probability distribution for the frequency of wind speed occurrence were not obtained by assigning uncertainty to the fundamental underlying parameters.

IPPSS report Table II-1 and the description of the structures at the Indian Point site are not pertinent to the analysis documented in this section. The wind hazard curves which were developed are for the Indian Point site and not for any specific structure or pairs of structures. It was learned at the meeting with PLG that a draft report for this section contained hazard curves for individual structures. Subsequently, curves were presented for The Indian Point site in general. Evidently, the discussion concerning plant and target definition is left over from the draft report.

Our understanding is that IPPSS report Figure II-1 outlines an area which was used to establish upper- and lower-bound hazard curves for tornado effects. We believe a more natural selection of a target area should also include structures from Unit 3. The effects of a large area would be to lower the lower bound and raise the upper-bound hazard curves slightly. However, since significant tornado strike areas are large relative to the site area overall effects of this difference are judged to be small to moderate.

SECTION III. TORNADO WIND SPEED RISK ANALYSIS

General

An independent check of the tornado wind speed hazard curves was conducted to confirm that the median IPPSS curve is reasonable. An approximate analysis was performed to verify the frequency value of 1.1×10^{-4} per year for tornadoes of any size hitting the site (see IPPSS Figure III-7) and to verify the frequency distribution of wind speeds given a tornado strike (i.e., the shape of the median hazard curve in IPPSS Figure III-7).

The frequency of occurrence of tornado strike was obtained using a mean rate of 2.425×10^{-4}/sq. mi/yr. from IPPSS report Table III-13 (note: as discussed below, we believe that this value is conservative), distribution of F-scale values from IPPSS report Table III-11, and distribution of tornado lengths and widths from Appendix B of IPPSS report, Section 7.9.5. Average tornado lengths and widths were calculated, and an average tornado origin area was computed to be equal to the sum of the average areas for the F-scale values weighted by the frequency of F-scale occurrence. This value times the mean rate of tornado occurrence produced a strike frequency of 1.2×10^{-4} /sq mi/yr which compares closely to the reported value of 1.1×10^{-4}/sq mi/yr. The approximate value assumes that all velocities in the tornado area are effective and that dependencies between F-scale value, tornado lengths, and tornado widths do not exist.

Two calculations were made to verify the distribution of wind speeds given a tornado strike. In the first check, only wind speeds corresponding to the median wind speed intervals from IPPSS report Table III-12 coupled with the distribution of F-scale values from IPPSS report Table II-11 were used. The velocity distribution was combined with the mean rate of tornado strike frequency (i.e., 1.1×10^{-4}/sq mi/yr). Figure 4 shows the results superimposed on the reported curves. The approximate analysis gives conservative values. This was expected since the reduction of origin area at higher wind speed was not incorporated into the calculation. In a second analysis, the origin area was reduced using average path lengths and path widths with frequencies corresponding to F-scale values obtained from Appendix B of IPPSS report Section 7.9.5. Figure 5 shows the results superimposed on the reported curves.

Based on these approximate calculations, we believe that the median curve has been rationally developed. As discussed below, we feel that the mean rate of occurrence used in the analysis is high and that the tornado hazard curves as a whole are conservative.

SECTION A. Methodology

The transmission line system component was not used in the analyses documented in the IPPSS report. Also, no tornado wind loads or effects of missiles are given in this report.

SECTION 1. Tornado Risk Model

We agree with the assumptions in this section.

SECTION 2. Tornado-Target Strike Model

We agree with the expression for a union definition of tornado-target interaction.

SECTION a. Target Intersection Damage Events

We agree with the expressions for an intersection and point source definitions of tornado-target interaction.

SECTION b. Transmission Line Targets

Hazard curves based on tornado-transmission line target interaction are not used in the IPPSS report; hence, this section was not reviewed.

SECTION 3. Tornado Windfield Model

The windfield model was not reviewed in detail. The results of the approximate analysis verified that the velocity distribution produced a secondary effect on the resulting hazard curves. The model used in the IPPSS gives reasonable results.

SECTION 4. Path Length Intensity Variations

One difference between the approximate calculation performed to verify the tornado hazard curves is that no path length adjustment was made. The close comparison with the results in the IPPSS report as shown in Figure 4 suggests that using the entire path length may not always greatly overpredict the probabilities of wind speed exceedance.

SECTION 5. Probabilistic Model of Tornado Data

No comment for this section.

SECTION 6. Simulation Methodology

No comment for this section.

SECTION B. Analysis of Tornado Data Record

Justification should be provided in the IPPSS report that 29 years of data (from the NSSFC record) is adequate to develop hazard curves which give the frequency of tornado strike on the order of 10^{-4} per year.

SECTION 1. Site Regionalizations

We agree that it is reasonable to study various regions surrounding the site in order to assess the variability of tornado risk for Indian Point.

SECTION 2. Prior Analysis and Selection of Tornado Population

In reviewing the statistical analysis given in Table III-4, we see no reason to favor the subregion over the 1-degree, 2-degree, or 5-degree areas. As an independent check, we reviewed the tornado statistics presented in Reference 1 which give maps of the United States that show numbers of tornado and path lengths are shown. A grid of numbers from the DAPPLE data base (from 1916 to 1977) shows the trend of tornado occurrence with location. There is a definite decrease in tornado activity at the Indian Point site as compared to a 5-degree area surrounding the site. A visual comparison indicates that the tornado rate at the site is less than half the rate based on a 5-degree area, and no F4 or F5 events are recorded at the site. This suggests that the tornado hazard at Indian Point based on the subregi^ is conservative.

We do not feel that the confidence bounds, which are provided, are meaningful. This bound assumes that the subregion area is correct and only reflects possible error in the tornado count. We believe that more useful bounds should include the effects of local conditions.

We accept the path length and path width distributions as reasonable.

SECTION 3. Adjustments and Error Analysis

We agree that data adjustments and error correction is worthwhile. However, the effect of this type of pot-ntial bias is overshadowed by the uncertainty in the statistics which are applicable to the Indian Point site.

SECTION C. Tornado Wind Speed Risk

Based on an approximate analyses and our belief that the tornado hazard at the Indian Point site is lower than the area selected in the IPPSS study, we feel that the median hazard curve shown in IPPSS report Figure III-7 is conservative.

The bounds on the curves were assumed to represent the 5th and 95th wind speed percentiles. No attempt was made in the analysis to rationally propagate the uncertainties in the individual problem parameters through the analysis. The procedure used is inconsistent with the approach used to develop the probability distribution of seismic hazard curves. In developing the seismic hazard curves, probabilities were assigned to maximum magnitude cutoff and source zones. Hazard curves were then systematically developed and the probabilities rigorously obtained. In addition, at the meeting with PLG, no evidence was available to suggest that the procedure used to establish the tornado hazard bounds could be verified by other studies conducted both rigorously and using this approximate approach.

Based on these observations, we do not believe that the two bounds or their associated 5-curve counterparts developed in Chapter VI are credible. In terms of offsite consequences, we believe that tornado effects do not dominate. Thus, the questions of whether the bounding curves are reasonable is not important for the IPPSS.

SECTION IV. Hurricane Windspeed Risk Analysis

The review of Section IV was performed by Dr. Larry R. Russell. His comments on this section are given in Appendix C. His basic conclusion is that the median hurricane hazard curve is unconservative because the IPPSS analysis did not consider severe topographic conditions for certain wind directions. We also have reviewed the results given in NBS Building Science Series 124 and find that this reference gives results in excess of the IPPSS median hurricane hazard curve.

In regards to the probability distribution of hazard curves, the IPPSS approach as documented in this section is identical to the approach used for tornadoes. We do not believe that the two bounding curves or their associated 5-curve counterparts developed in Chapter VI are credible.

SECTION V. EXTRATROPICAL CYLONE AND THUNDERSTORM RISK

The hazard values obtained for extratropical cyclones and thunderstorms were checked against References 2 and 3. The values we obtained compared reasonably well with the curves in the IPPSS. Reference 3 which is the more recent report, indicates that the wind speed values may be low by approximately 10 percent.

SECTION VI. WIND SPEED PROBABILITY FUNCTIONS

The procedure used to develop the 5 hazard curves for each wind type is rational and reasonable. However, as explained previously, we do not believe that the bounds for hurricane or tornados are meaningful. Hence, the two lower and two upper bounds created for each wind type also are not meaningful. This is not a problem for tornado effects since this wind type ultimately is not a major contributor. This is not the case for the effects of hurricane and extratropical cyclones. As stated for Section IV, we recommend that a set of curves be developed which rationally propagates parameter uncertainties through the analysis leading to a family of hazard curves each with a associated probability value.

We noted an error in Table VI-1. The value for the median curve at 200 mph is 2.6×10^{-8} per year which appears to be low by an order of magnitude.

SECTION 8.3.4 <u>IDENTIFICATION OF MAJOR SCENARIOS, SYSTEMS, AND STRUCTURES CONTRIBUTING TO RISK - INDIAN POINT 2</u>

Scope of Review

This section of the IPPSS summarizes the major scenarios, systems, and structures/equipment which contribute to the risk of the various consequences. We offer general comments here concerning the statements made and specific comments where appropriate. More comprehensive comments are made earlier in this report for the other sections of the IPPSS.

SECTION 8.3.4.1 Seismic

Although the median capacity of the city water tank has been downgraded to 0.25g since the IPPSS was published, the failure of both the RWST and the condensate storage tank are required for core melt. Other components in series which individually could cause failure (i.e. cable trays, containment shear wall, and diesel generator fuel oil tanks) have only slightly higher capacity than the RWST and condensate storage tank. These components also would be significant contributors if item 2 did not fail. In addition, if the cable tray supports are considered to be independent (see discussion for IPPSS Section 7.2.4) this component would likely become the next most important contributor to frequency of offsite consequences after item 2. Because we believe that the D&M hazard curves are more representative of the Indian Point site, the frequency of offsite consequences due to seismic is a factor of 2 higher.

SECTION 8.3.4.2 Wind

We agree with the description of the various contributors. As discussed for Section 7.5.1 we believe that offsite power will fail at a wind velocity much lower than the median capacity of 140 mph postulated in the study. In addition, we believe that the hurricane hazard curves are unconservative. This will raise the frequency of total loss of AC power and hence 2RW release by a factor of 20 for the effects of wind.

SECTION 8.3.5 IDENTIFICATION OF MAJOR SCENARIOS, SYSTEMS, AND STRUCTURES CONTRIBUTING TO RISK - INDIAN POINT 3

SECTION 8.3.5.1 Seismic

We agree with the description of the major seismic core melt scenarios. It should be noted that failure of the diesel generator fuel supply and the control building shear wall dominate the core melt fragility with an equivalent median failure value of about 0.8g. As discussed for Section 7.2.5.3 we believe that the capacity of the hung ceiling in the control room is less than this value (failure of the ceiling could incapacitate the operators). In addition, we believe the seismic hazard is a factor of 2 higher. We judge that the mean frequency of core melt and offsite consequences could be 2 to 10 times larger than given in the IPPSS.

SECTION 8.3.5.2 Wind

Even using the median fragility wind velocity value of 140 mph for offsite power, the logic for 2RW release is dominated by only the service water pumps, since the power is more likely to fail. As discussed for Section 7.5.3 we believe that offsite power will be lost at a much lower wind velocity than 140 mph. However, since the tornado hazard curves are conservative, we judge that the mean frequency of core melt and offsite consequences for wind are reasonable.

TABLE 1
(Ref. 6)

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. Standard Project Flood & Ashokan Dam 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. This reach extends from the Battery to the Tappan Zee Bridge, about mile point 27. Actual flow at Indian Point, some 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.

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TABLE 2.

SUMMARY OF EQUIPMENT CAPACITY VALUES

Section	Component	Median Ground Acceleration (g) Unit 2	Unit 3
5.2.1.1	Reactor Pressure Vessel	3.80	3.80
5.2.1.2	*RPV internals	1.04	1.04
5.2.1.3	Steam Generator	1.84	1.84
5.2.1.4	Reactor Coolant Pump	3.04	3.04
5.2.1.5	*Pressurizer	0.87	0.87
5.2.1.6	Control Rod Mechanism	2.84	2.84
5.2.1.7	Reactor Coolant Piping	6.16 to 8.85	5.59 to 17.70
5.2.1.8	Safety Injection Pump	2.40	2.17
5.2.1.9	Residual Heat Exchanger	10.56	10.56
5.2.1.10	Component Cooling Water Heat Enchanger	5.43	6.13
5.2.1.11	Accumulator Tanks	13.71	15.37
5.2.1.12	Boron Injection Tank	2.53	4.95
5.2.2.1	*Containment Fan Coolers	1.16	2.17
5.2.2.2	*Residual Heat Removal Pumps	1.70	1.70
5.2.3.1	*Piping and Supports	1.40 to 12.45	1.40 to 17.70
5.2.3.2	Generic Equipment Structural Mode		
	*Diesel Oil Storage Tank	1.14	1.14
	Service Water Pumps	2.47	2.47
	*RWST	0.70	0.70
	*Condensate Storage Tank	1.28	1.28
	*Ductwork and Dampers	1.12	1.12
	*Transformer	1.69	1.69
	*Relief Tank	1.37	1.37
	*Batteries and Rack	1.37	1.07 to 1.29

*Key Equipment: From Tables 7.2-3 and 7.2-7 of IPPSS Report

TABLE 2.

SUMMARY OF EQUIPMENT CAPACITY VALUES (continued)

Section	Component	Median Ground Acceleration (g) Unit 2	Unit 3
5.2.4	Capacities Derived from Tests (Reactor Protection Systems)	4.75	5.07
5.2.5	*Generic Electromechanical and Electric and Control Equipment	1.30 to 2.49	1.17 to 1.51
5.2.6	*Generic Capacities for Values (Motor and Air Operated)	3.17 to 5.11	3.17 to 5.11
5.2.7	*Cable Trays	1.10	2.20
5.2.8	*Offsite Power	0.2	0.2

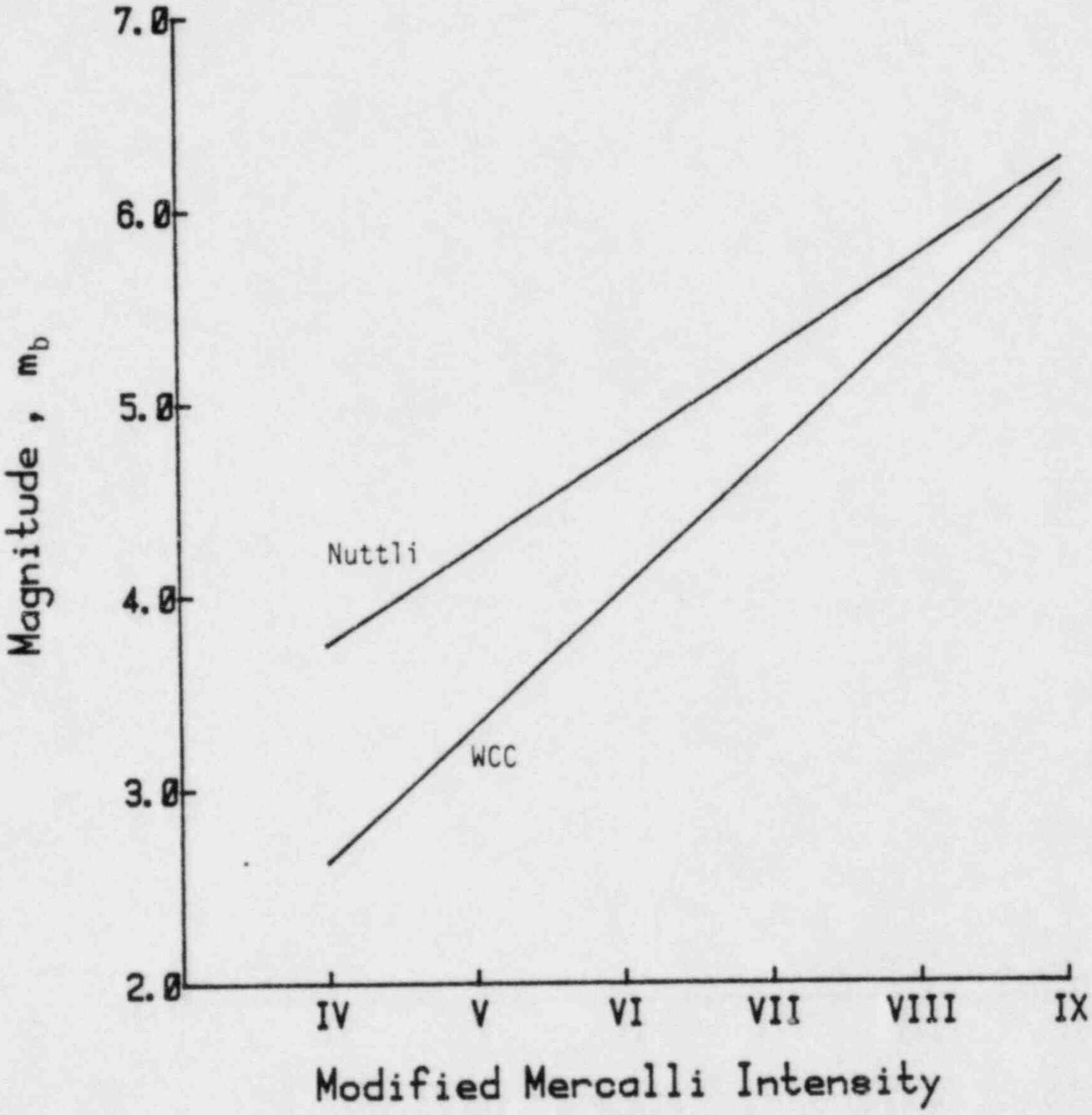


Figure 1. A comparison between WCC I-M_b relation and Nuttli's curve.

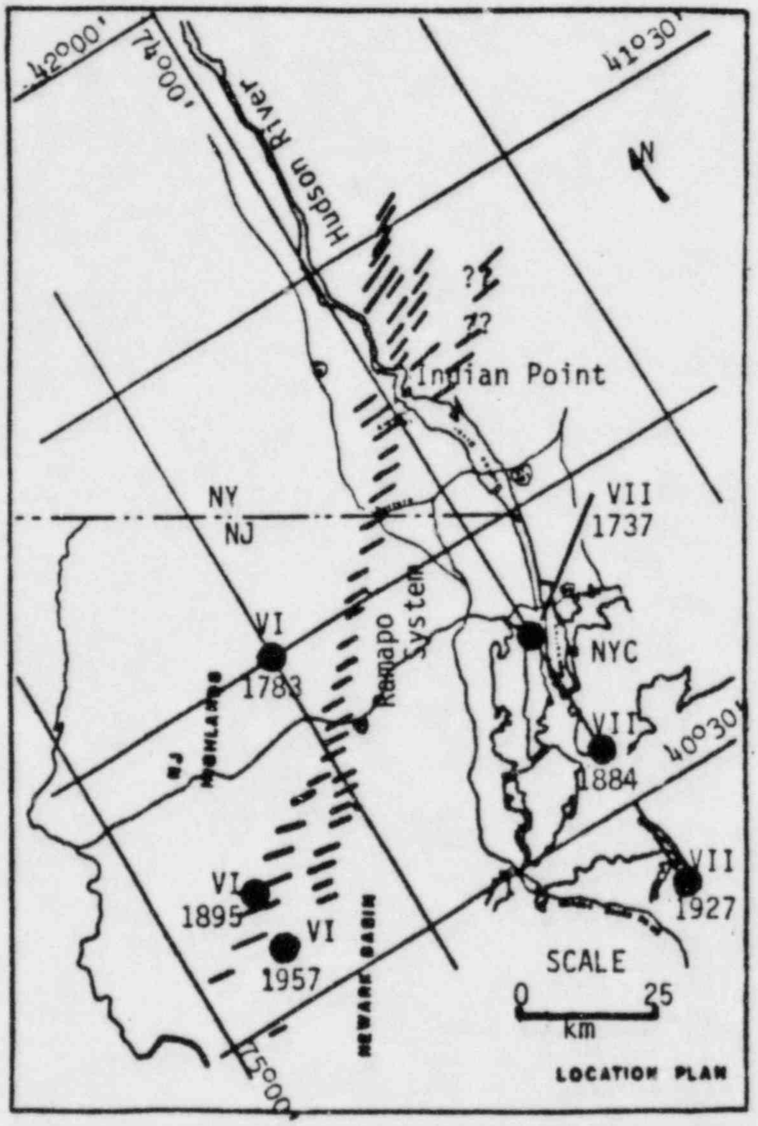
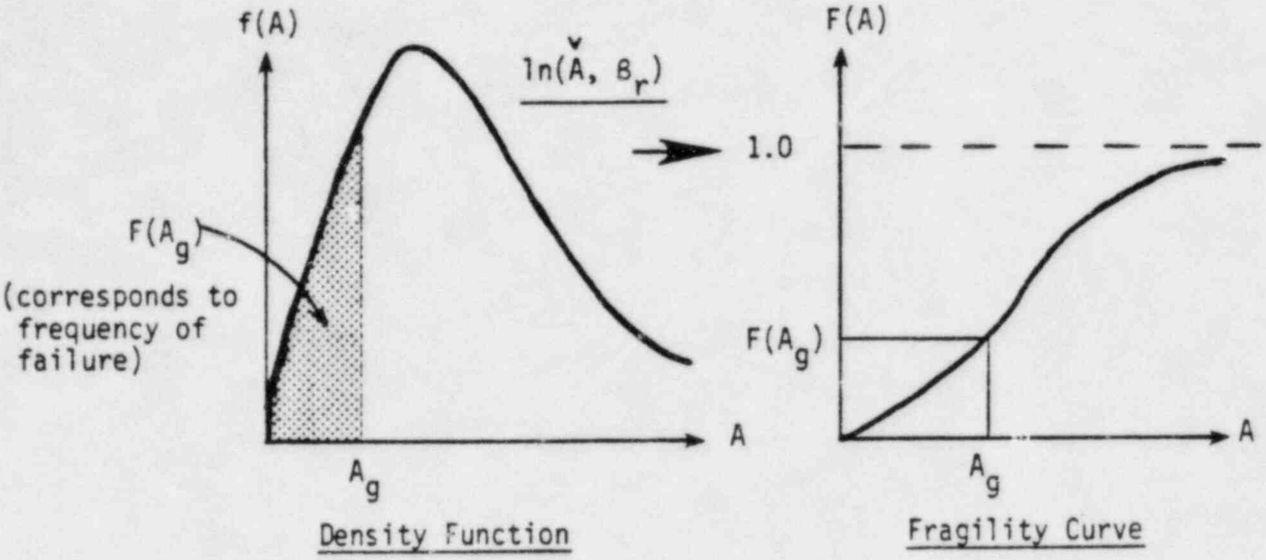
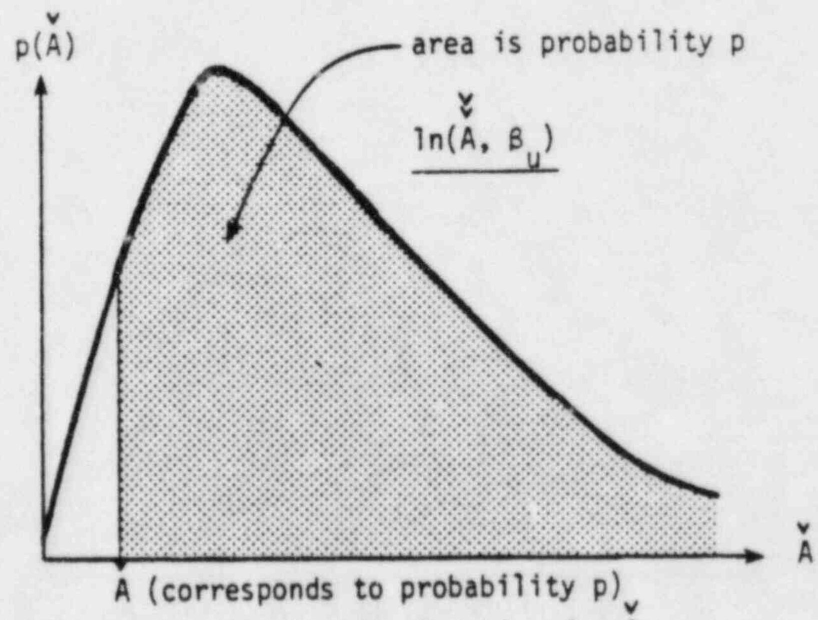


Figure 2. A map indicating the location of the largest events near the Indian Point site.

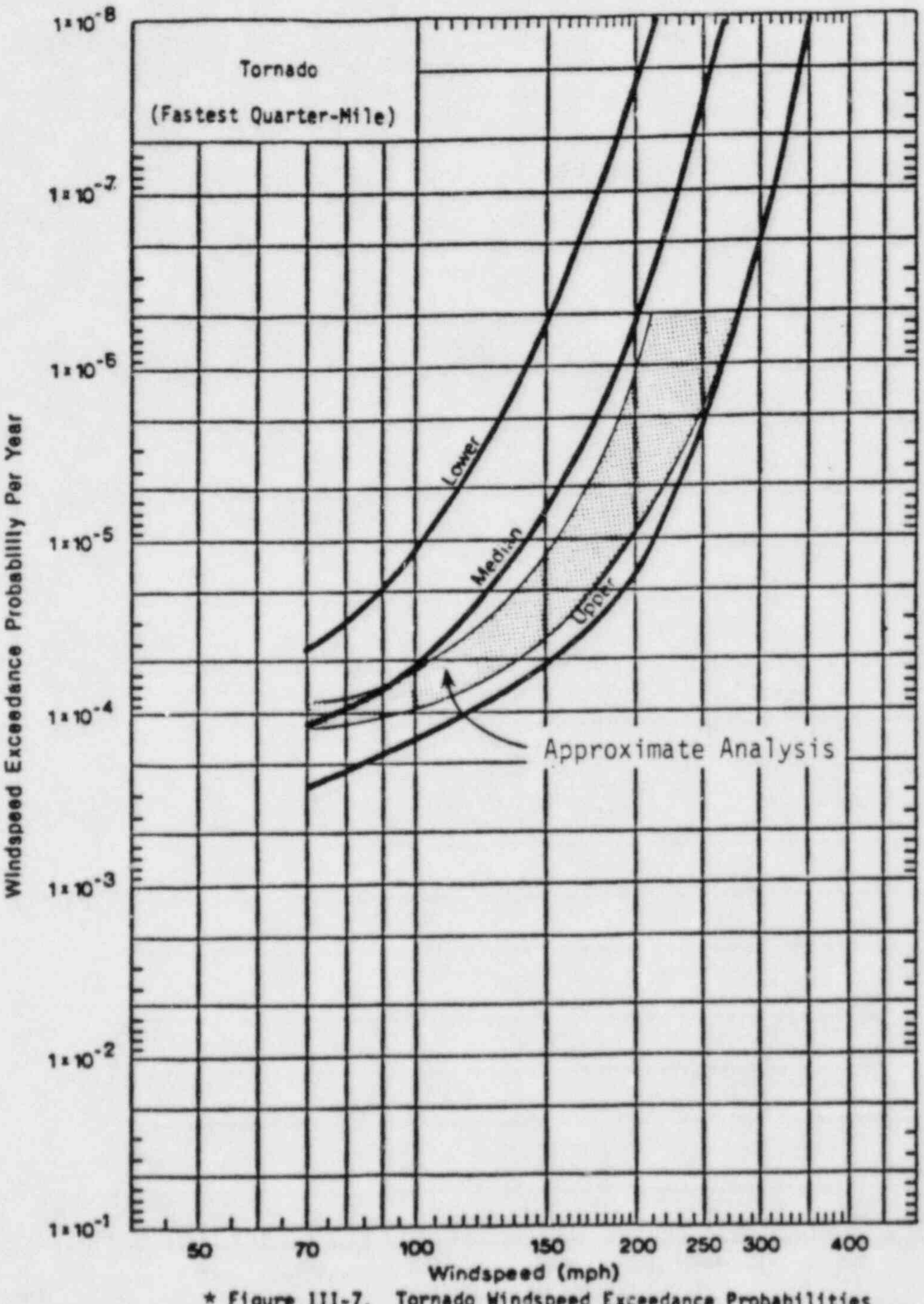


(a) Capacity Frequency Density Function and Fragility Curve



(b) Probability Density Function for $\check{A}$

Figure 3 Probability and Frequency Functions for Fragility Analysis



* **Figure III-7. Tornado Windspeed Exceedance Probabilities at 33 ft. Elevation**

* IPPAA Report, Section 7.9.5

Figure 4 Comparison of Approximate Tornado Analysis (without Origin Area) with IPPSS Results

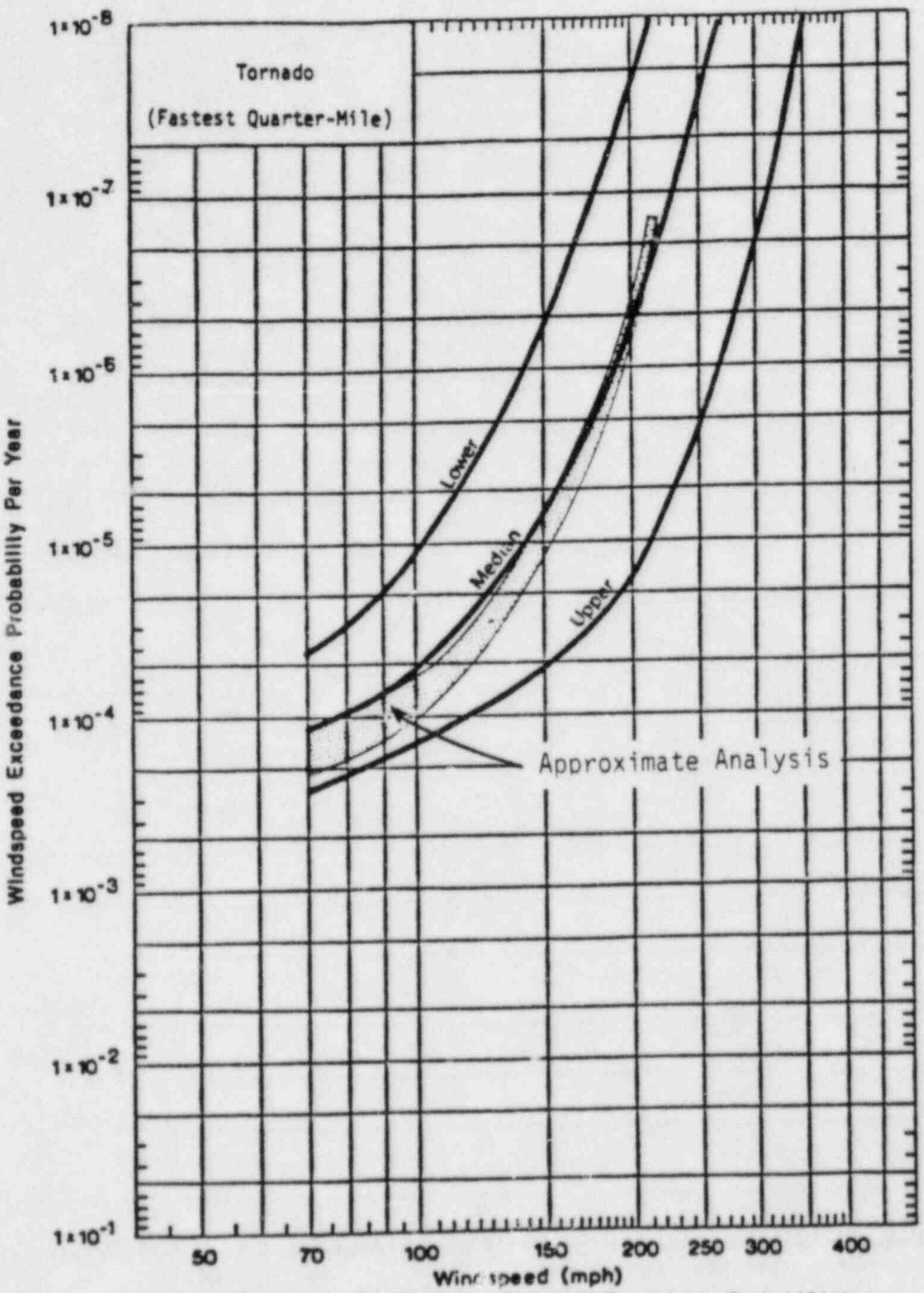


Figure III-7. Tornado Windspeed Exceedance Probabilities at 33 ft. Elevation

Figure 5 Comparison of Approximate Tornado Analysis (with Origin Area) with IPPSS Results

4. SEISMIC HAZARD ANALYSIS

In an effort to more fully understand the impact of various deficiencies that were pointed out in our review of the seismicity analyses conducted for the Indian Point Site, a study was undertaken to evaluate the sensitivity of key parameters and results. The objectives of this study are the following: to evaluate the sensitivity of the seismic hazard analysis to variation in key parameters; to check results presented in the seismicity analyses; to investigate alternative hypotheses not considered in the study; and to evaluate the implication of these alternatives relative to the mean frequency of release category 2RW. The intent here is not to conduct another independent seismicity study for the Indian Point plants; we instead investigate the impact on key results of particular deficiencies that were recognized as a result of our review. The references used in this study are listed at the end of the chapter.

SEISMIC HAZARD MODEL AND ANALYSIS

The seismic hazard model used in this study is similar to the ones used in the IPPSS. The model was originally proposed by Cornell (Ref. 1), and described with various improvements in References 2, 3, and 4. A standard computer program was used to conduct the analysis and is described in Reference 5.

The steps in the seismic hazard analysis are:

- Identification of seismic source zones based on historic seismicity and tectonic evidence.
- Estimation of seismicity parameters including upper bounds on event size.
- Selection of an attenuation model appropriate for the region of interest.

In the analysis it is assumed that the seismicity is distributed uniformly in a source zone. The distribution of earthquakes is described by the Gutenburg-Richter recurrence relationship, and their random occurrence is assumed to be spatially and temporarily independent. The uncertainty about the attenuation curves is assumed to be described by a lognormal distribution. These aspects of the analysis are consistent with the methods used in the IPPSS.

Check of the Dames and Moore - Woodward Clyde Seismicity Curves

On the basis of information provided in the IPPSS seismicity studies on seismicity parameters, source zone geometry, and attentuation models, a check of a few of the seismicity curves was made. Our intent was to verify the accuracy of the analyses performed.

For the Dames and Moore study, the Piedmont zone was selected. The seismicity curves for two alternative hypotheses were checked; these were the best estimate case for b=0.90 and $M_{b\ max}$=5.70, and the case of b=0.76 and $M_{b\ max}$=6.0. Our results in the acceleration range 0.10 to 0.70g are within 30 percent of those calculated by Dames and Moore.

For the Woodward-Clyde seismicity study, results for Source 1 were checked for two maximum event sizes, intensity VII and VIII and their preferred attenuation model. Our results were within 50 percent in each case in the acceleration of 0.10 to 0.80g.

Sensitivity Analysis

In each of the two seismicity studies for the Indian Point site, a sensitivity analysis was conducted to demonstrate the effect on the seismicity curves to variations in key parameters or model selections. For those parameters investigated, the impact was well demonstrated. However, as noted in our review comments in Section 7.2, the variability in ground motion predictions is a function of the path taken to make the estimate. The total uncertainty in ground motion predictions can, as demonstrated in Reference 6, increase as a result of transformations made to reach the final variable of interest. This is a source of modeling uncertainty not considered in the IPPSS.

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We consider the effect on the seismicity curves to variations in the logarithmic standard deviation of the distribution about the attenuation curve. Recall that in the IPPSS a value of 0.60 was used for $\sigma_{\ln a}$. This value is typical of the scatter in strong motion data. No variability in this parameter was considered in either of the seismicity studies. The variation in seismicity curves is considered for three values of $\sigma_{\ln a}$, 0.60, 0.70, and 0.80, which correspond to factors of 1.80, 2.01, 2.23 uncertainty in ground motion at the one standard deviation level. Figure 1 presents the results for the three values of $\sigma_{\ln a}$. The source used in this particular example is similar to the Ramapo fault zone that will be considered later in this chapter. The results indicate that variation in the frequency of exceedance curves is less than a factor of 3 for sustained accelerations in the range 0.10 to 0.70g.

In the WCC study and in Section 7.2, the seismicity curves have been truncated to reflect the belief that the accelerations are limited. This truncation was made outside the hazard calculation, by simply limiting the range of the derived hazard curves developed with no truncation. It was suggested at the PLG meeting that this truncation should have been performed within the analysis. Although generally agreed that this was the more correct way to carry out the truncation, the procedure of truncating the distribution aposteriori is conservative in that the frequencies of exceedance for accelerations less than the truncation point will be higher than if the truncation had been performed in the probabilistic analysis by properly truncating the distribution and normalizing to give unit area. This was verified for a simple example. To quantify this effect for the IPPSS in general, each seismicity curve would have to be recomputed. This has not been done.

Investigation of Alternative Modeling Hypotheses

The Dames and Moore and Woodward Clyde seismicity studies were judged to be inadequate for the reason that a Ramapo fault zone was not accounted for (i.e., they were given a zero probability weight). The basis of this judgment is the fact that considerable scientific investigation has focused on the Ramapo fault (Ref. 7, 8, 9, and 10). However, conclusions on the subject of the Ramapo fault as an earthquake generating source vary. These efforts to understand the source of seismic activity in and around the Ramapo fault zone are clouded with considerable uncertainty, making attempts to define the geometry of a source zone and seismicity parameters to go along with it difficult. Geologic investigations further tell us that the geologic framework of this region is very complex. Historical seismicity in the region exhibits considerable spatial scatter that fails to clearly delineate a Ramapo fault zone.

However, the deployment of a dense array seismic of stations in recent years has yielded a pattern of low magnitude events ($M_b \le 3.0$) around the Ramapo fault. Seismological studies of this data (Ref. 7 and 10) has led to a hypothesis that the Ramapo fault is a reactivated fault that is currently generating low magnitude events on a well defined surface.

This brief summary suggests that although considerable uncertainly exists as to the character of the Ramapo fault as an earthquake generating source, enough evidence is available to warrant its consideration in a family of possible source zones. Recall that an objective of a probablistic risk assessment is to consider reasonable hypotheses, in a manner consistent with the state of information and our degree of belief. As such, the considerable uncertainty in the source geometry and in seismicity parameters may warrant assigning of a low (nonzero) probability weight to this particular hypothesis.

In this section we consider a Ramapo fault zone and evaluate seismicity curves for various parameter alternatives. Our analysis utilizes the work in References 7 and 8 to develop seismicity parameters and source geometry models. As part of our analysis, we will not assign a probability weight to a set of seismicity curves corresponding to a Ramapo fault zone. We will, however, consider the effect of a Ramapo source as a function of an assigned probability weight. This will be addressed in the next section.

Figure 2 shows the Ramapo fault and the region near the Indian Point site. Also shown on the map is the location of a number of the largest events (MMI ≥ VI) that have occurred in the area.

As we mentioned previously, considerable uncertainty exists about the geometry of a fault zone to be utilized in a hazard analysis, and in values for the seismicity parameters. We choose in this analysis to take Reference 7 and 8 as a guide to our parameter selection. The following variables are considered in a sensitivity analysis:

- source geometry
- activity rate
- b-values
- upper-bound magnitudes

The attenuation model used in this study is Nuttli's relation for sustained acceleration. The logarithmic standard deviation for the distribution about the attenuation curve is taken as 0.60. A cutoff is applied to this distribution at the $5\sigma_{\ln a}$ level. This corresponds to a factor of 20 times the median.

We consider the Ramapo fault zone suggested in by Aggarwal and Sykes (Ref. 7) and their recurrence relationship for events greater than M_b=4 and a b-value of 0.73. This source is taken 15 km either side of the fault and is approximately 140 km long. This rather small source area represents a dense concentration of seismicity around the site. In these calculations no background seismicity was used. In Figure 3 the seismicity curve with an $M_{b\ max}$= 6.25 is shown with the curve developed by Dames and Moore for the Piedmont-Cape Ann zone. Differences in the two curves are within a factor of 3 out to about 0.70g. Also shown on the same figure are the seismicity curves for $M_{b\ max}$ values of 5.50 and 5.75. The figure reveals that a Ramapo fault zone does not significantly increase the hazard over results previously obtained for other source zones.

In Figure 4 we investigate the effect of varying the Richter b-value. Two cases are considered, b=0.73 and 0.90. The seismic activity rate is held the same, thus the effect is to considerably lower the occurrence rate of large events. As a result, the seismicity for a b=0.90 is considerably less (by an order of magnitude) for the case of b=0.73.

Variations in the seismic activity were considered; however, the effect is relatively small, as the frequency of exceedance is approximately linearly dependent on this variable. For this reason, we do not present specific results for variations in this parameter.

In Reference 8, Fischer presents different alternatives for a source zone size and recurrence relation. We consider the case of a Ramapo zone which is approximately 100 km square. A b-value approximately the same as the Aggarwal and Sykes value is used, and an activity rate based on a time period of 350 years used. The result for an $M_{b\ max}$ of 6.25 is compared to the results for the Aggarwal and Sykes source is shown in Figure 5. The impact of a larger source area, and longer time period for the data base has been to reduce the seismicity per year per unit area, resulting in lower frequencies of exceedance.

The above sensitivity analysis has not been an exhaustive survey of the full range of alternatives that might be considered to model a Ramapo fault zone. We have instead presented a range of possibilities suggested in the literature that we feel reasonably represents the range on the seismicity curves associated with a Ramapo source.

In order to understand the potential impact of considering a Ramapo fault zone in the risk analysis, we present a comparison of the mean frequency of release category 2RW due to seismic events to the frequencies computed in the IPPSS. The comparison is made in terms of the ratio of results calculated here for the Ramapo fault to the values we calculated using the original family of seismicity curves. Thus the rates are directly applicable to the results given in the IPPSS report. The comparison is made for Indian Point Units 2 and 3. The hazard curves used in the analysis were taken from IPPSS report Tables 7.2-4 and 7.2-8 for Unit 2 and 3, respectively.

The results of the comparison are given in Table 1. The implication of these results and the manner in which they should be viewed is discussed in the next section.

IMPLICATION OF A RAMAPO FAULT ZONE

The increase in the mean frequency of release category 2RW due to different representations of a Ramapo fault zone are presented in the previous section. The results given there show the increase due to the Ramapo source in comparison to mean frequency values obtained in the IPPSS. The next step is to postulate, in a Bayesian sense, a subjective weight for the Ramapo source and then combine the effect with the other postulated sources. Based on the information we have to date, we are unable to make a formal assignment for the Ramapo source. However, we have investigated the implication of various weights which could be assigned. At one limit is the probability assigment of 0. This implies that the Ramapo source is incapable and thus cannot possibly occur. At the other extreme is the probability assignment of 1.0 which says that the Ramapo source, plus a reasonable background seimicity which was added, replaces the other source zones considered in the IPPSS. This is obviously a very conservative scenario since it is highly unlikely that the only possibility is the Ramapo zone. For purposes of this sensitivity analysis, the D&M Piedmont zone with a M5.7 maximum magnitude is selected to be the background seismicity. This is also conservative.

Because there is a difference in integration procedures used by PLG and us, we have normalized the increase in mean frequency of consequences to correspond to the values given in the IPPSS report. In this section we have not included any other differences which we found in our review. Thus the results presented here are given in addition to changes we noted elsewhere in this report.

Figure 6 shows the effect of the Ramapo fault zone and its assumed background seismicity on the mean frequency of core melt or release categories for subjective probability values between 0 and 1. The curves were developed for release category 2RW. However, we expect the trend to be similar for other release categories and for core melt as well. Curves given for both Unit 2 and Unit 3 represent the <u>ratio</u> of the total seismicity-caused mean frequency (including the weighted contribution from the Ramapo source and background seismicity) to the seismically-caused mean frequency values corresponding to the IPPSS report (i.e., 1.4×10^{4} per year for Unit 2 and 2.4×10^{-6} per year for Unit 3). Thus the results shown in Figure 6 pertain only to seimically-caused consequences.

The two curves for each plant shown represent lower and upper bound possible Ramapo fault zones. These correspond to hazard curves 3 and 1 given in Table 1, respectively, which are discussed in the previous report section.

Figures 7 and 8 show similar plots for total mean frequency of release category 2RW and core melt, respectively. In these plots the mean frequency values given in IPPSS report Tables 8.3-2 and 8.3-3 were used as the base values for Unit 2 and Unit 3, repectively. Thus the effect of the Ramapo fault zone on higher level consequences as function of the subjective probability for the zone can be seen.

By comparing Figures 6, 7, and 8, it is seen that the effect of the Ramapo Fault zone decreases monotonically from seismic-caused release categories, to total release category 2RW, and finally to core melt. The reason the effect of the Ramapo decreases is because other events such as fire, hurricane, tornado, and internal accidents dilute the contribution made by the Ramapo source.

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11. Woodward - Clyde Consultants, " A Seismic Exposure Study For the Indian Point Nuclear Generating Station," 1981.

12. Dames and Moore, "Seismic Ground Motion Hazard at Indian Point Nuclear Power Plant Site," for Pickard, Lowe and Garrick, Inc., February 18, 1981.

TABLE 1

RATIOS OF MEAN FREQUENCY OF RELEASE CATEGORY 2RW

Source Zone	IP-2	IP-3
Aggarwal & Sykes (Ref. 7)		
1. b=0.73. $M_{b\ max}$ = 6.25	10.8	34
2. b=0.73, $M_{b\ max}$ = 5.75	6.4	10
3. b=0.73, $M_{b\ max}$ = 5.50	4.5	5.5
4. b=0.90, $M_{b\ max}$ = 6.25	1.5	4.1
Fischer (Ref. 8)		
5. b=0.70, $M_{b\ max}$ = 6.25	2.51	6.25

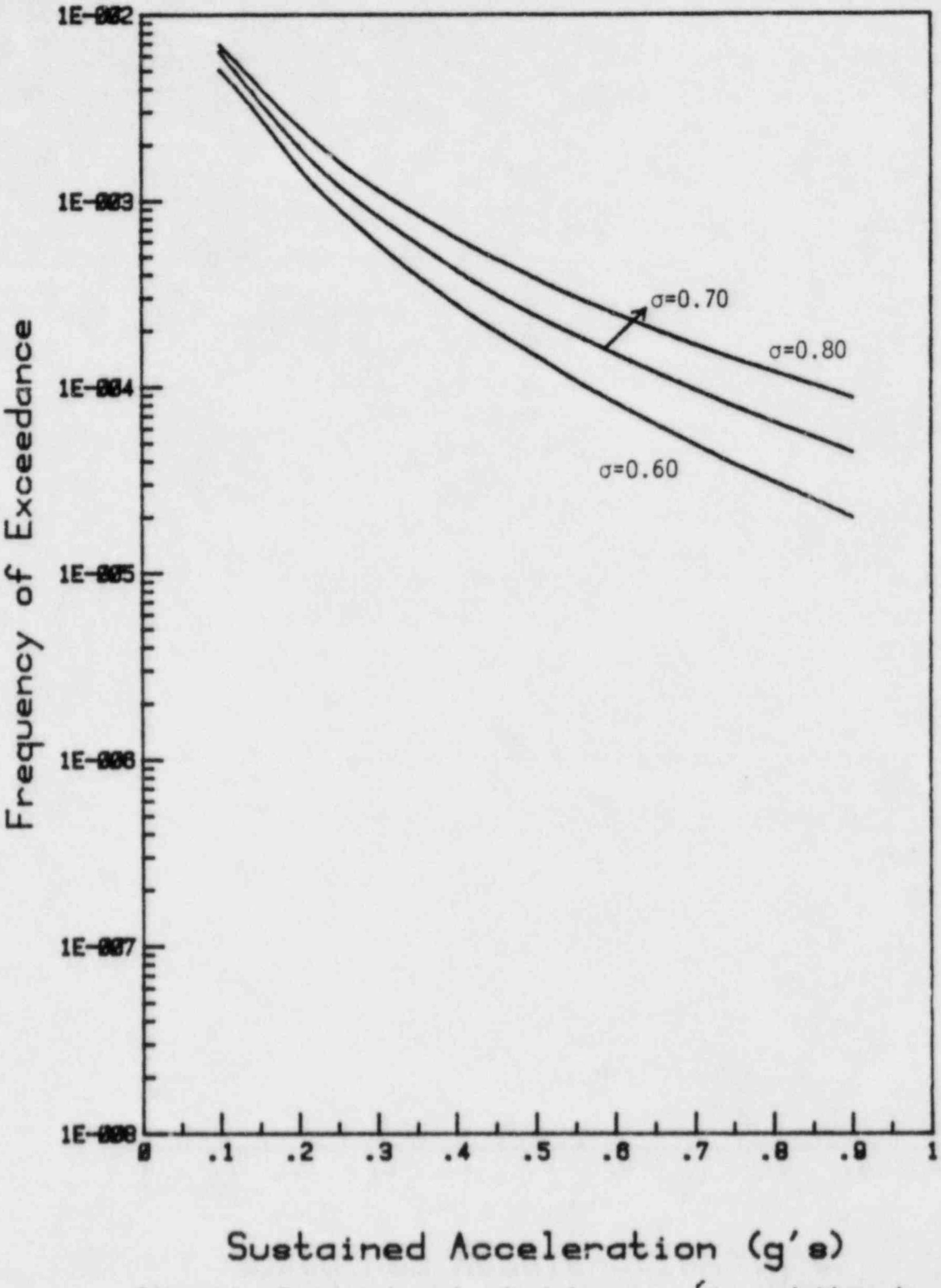


Figure 1. Sensitivity of seismicity curves to variations in the logarithmic standard deviation in the distribution about the attenuation curve. Results for three values of $\sigma_{\ln a}$ are shown, 0.60, 0.70, and 0.80.

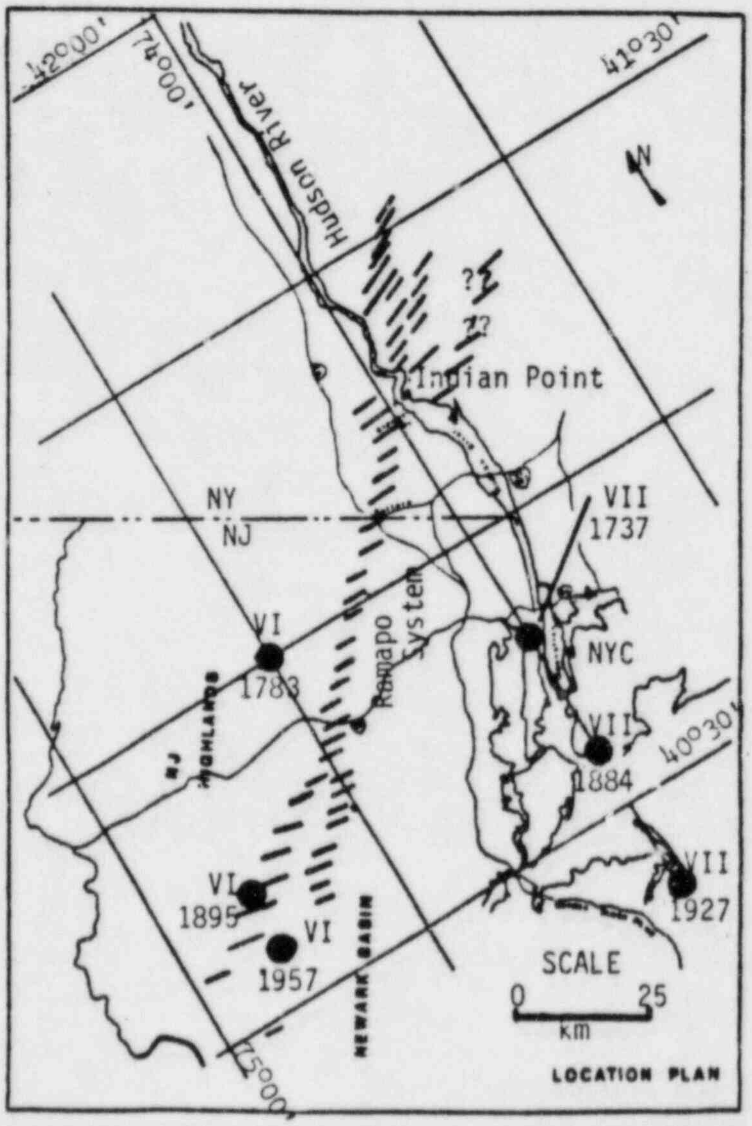


Figure 2. The Ramapo Fault and the region surrounding the Indian Point site (modified from Ref. 8).

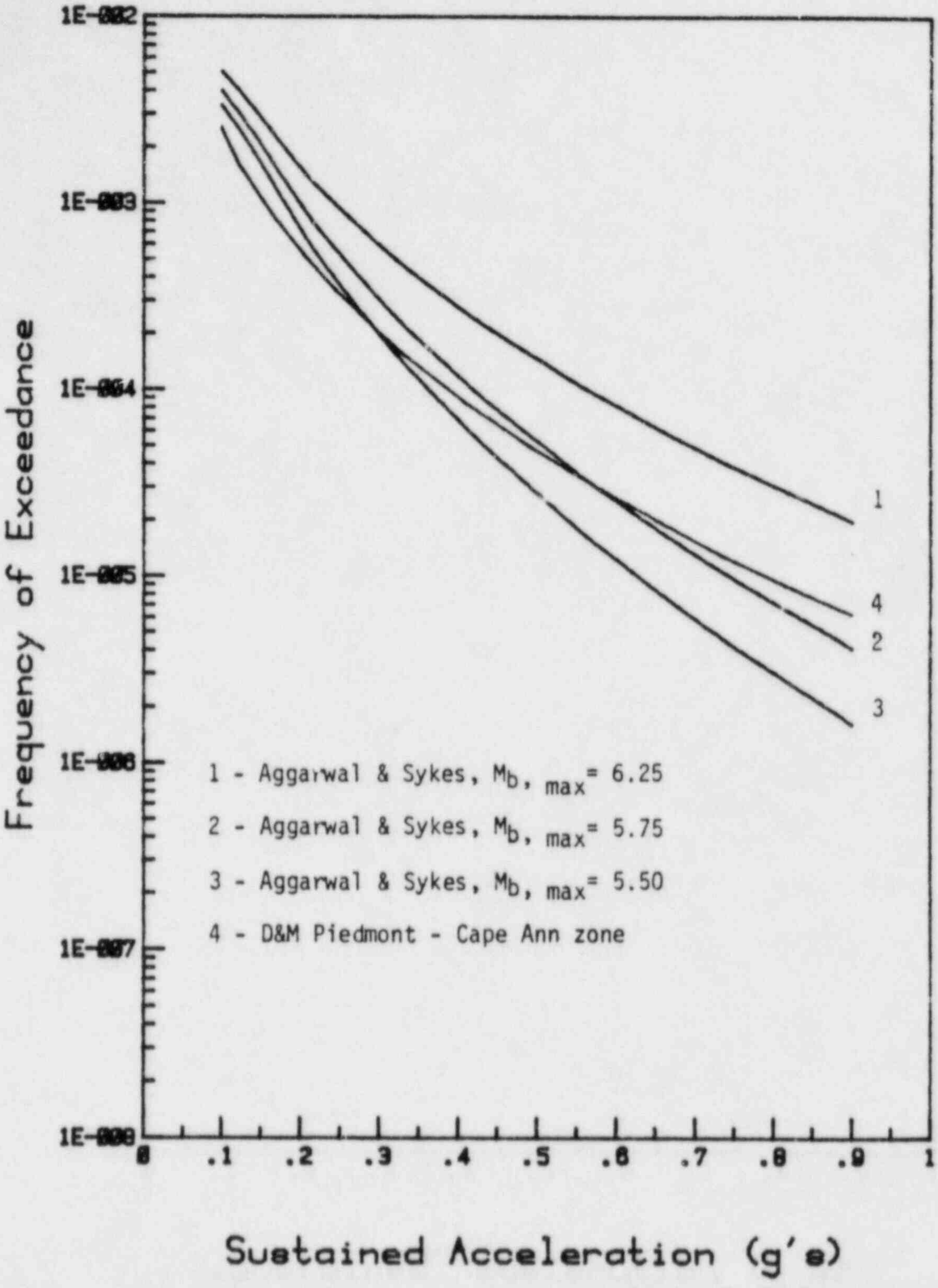


Figure 3. A seismicity curve for the Ramapo fault for $M_{b,\ max}$ values of 6.25, 5.75, and 5.50 and the recurrence relation suggested by Aggarwal & Sykes shown with the Piedmont-Cape Ann seismicity curve in the Dames and Moore Study.

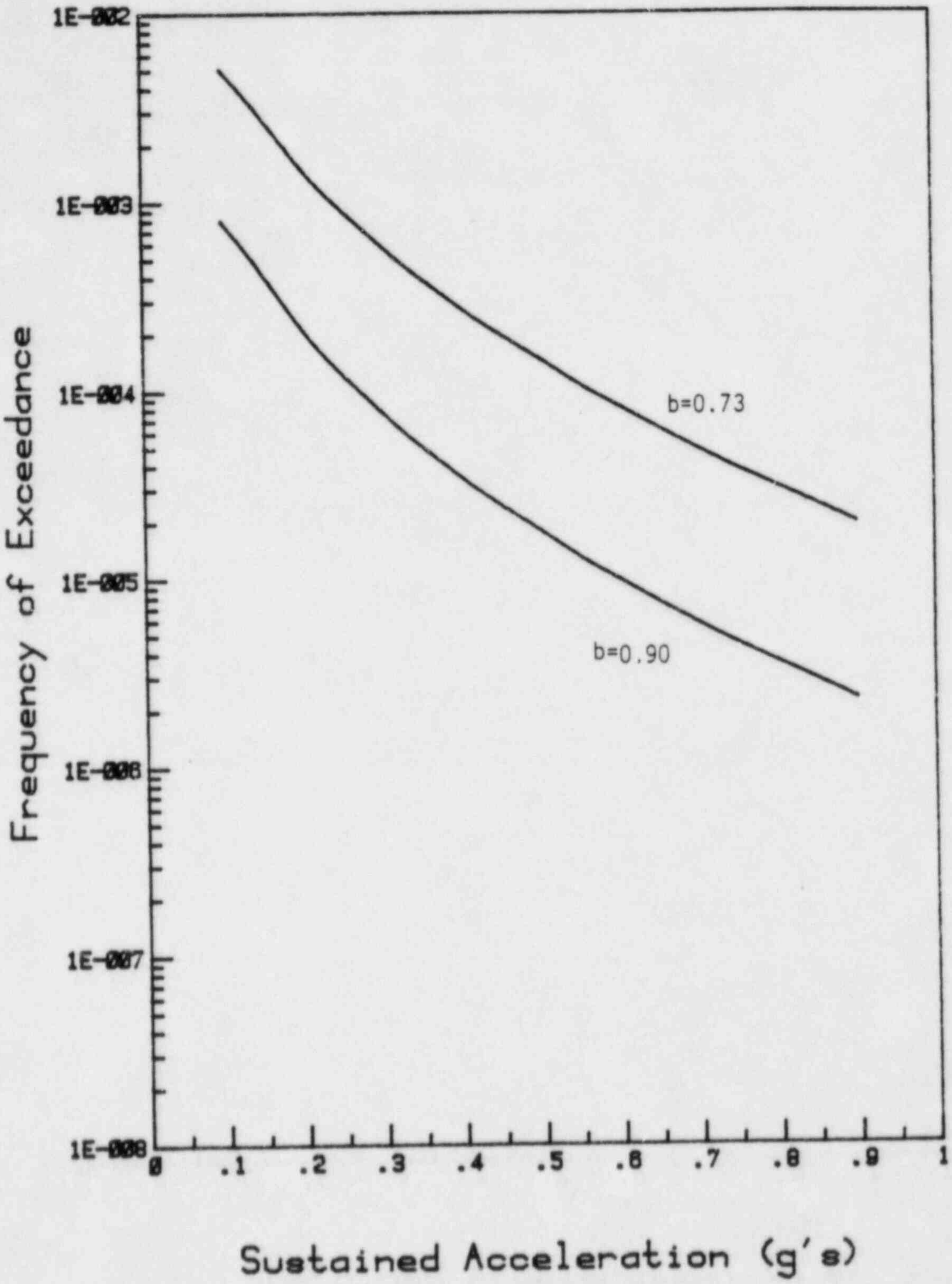


Figure 4. The effect of b-values on the seismicity curves for the Aggarwal and Sykes Ramapo fault zone.

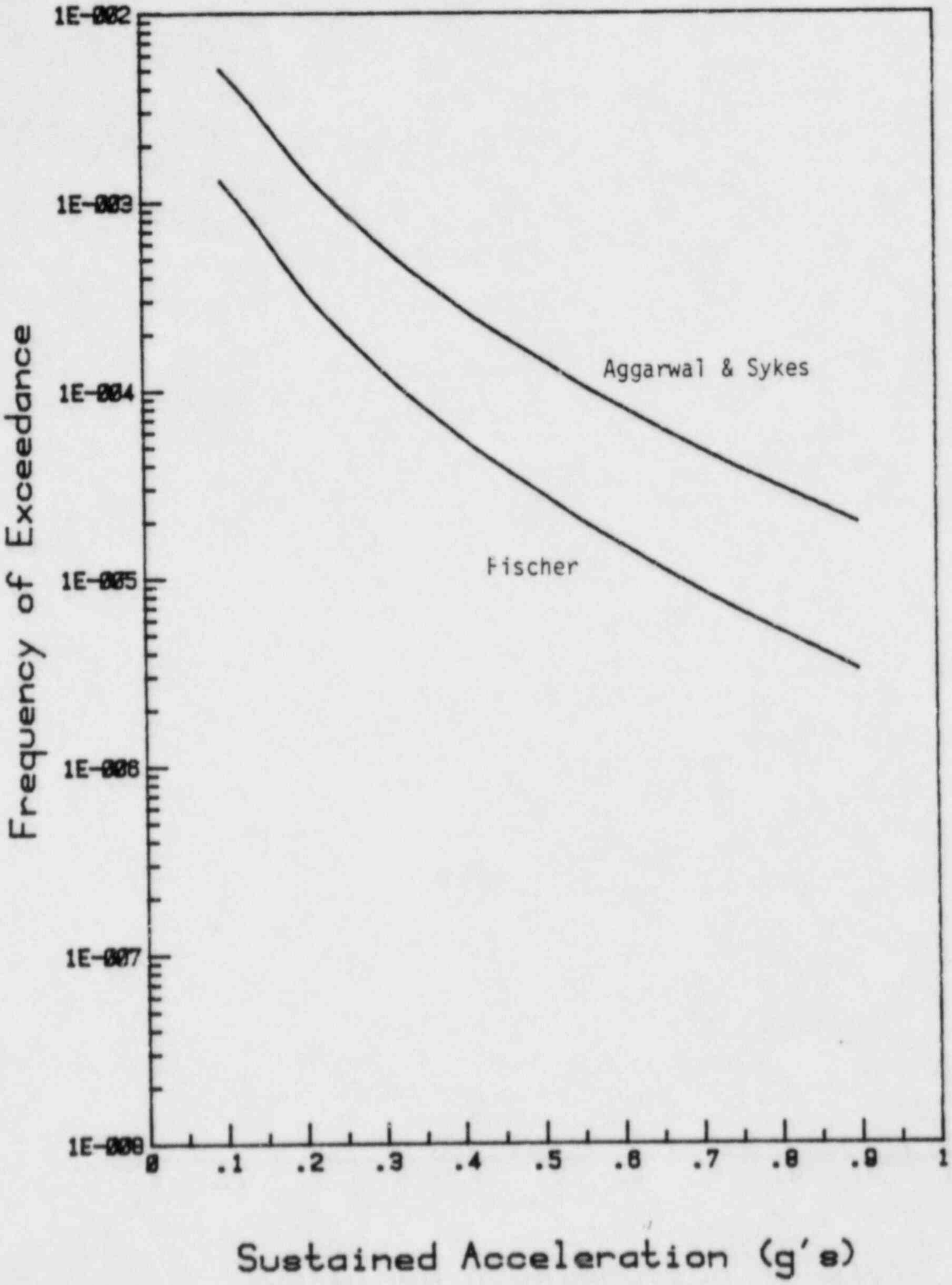
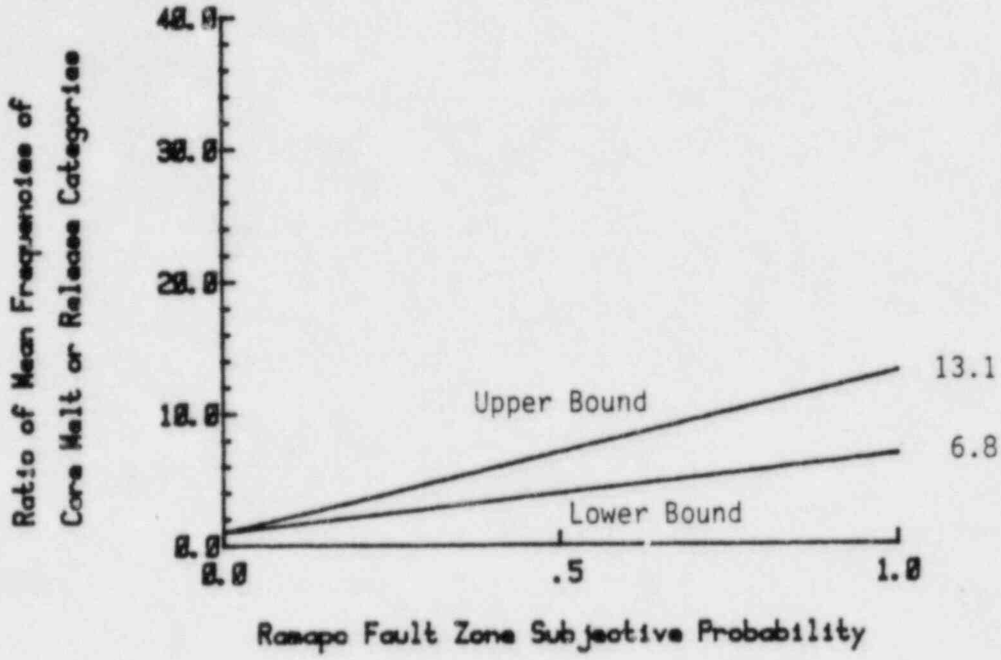
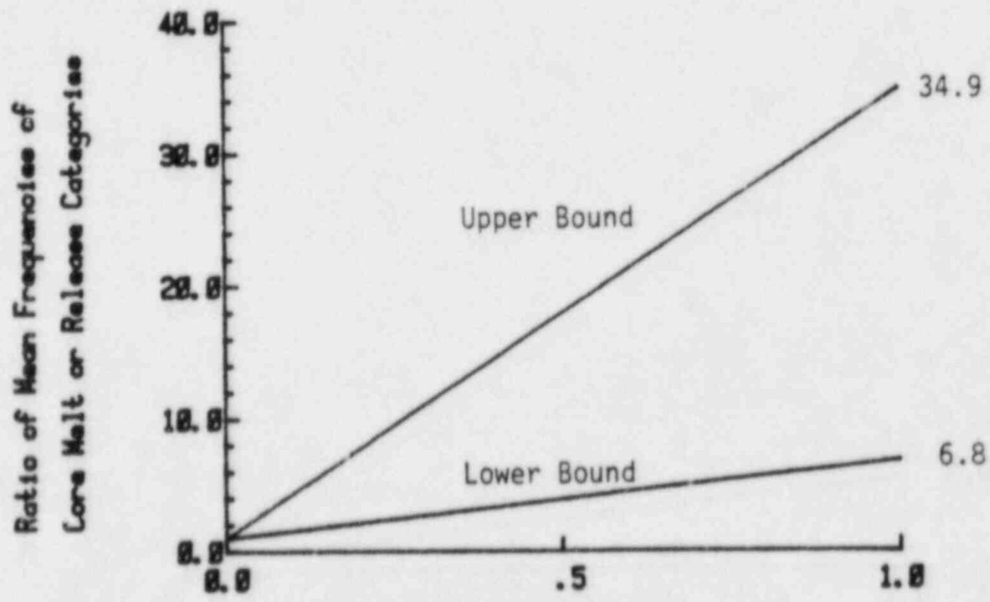


Figure 5. A comparison between the Aggarwal and Sykes (Ref. 7) source zone and a larger source area considered by Fischer (Ref. 9).



(a) Unit 2



Ramapo Fault Zone Subjective Probability

(b) Unit 3

Figure 6 Effect of Including a Ramapo Fault Zone on Seismic-Caused Consequences

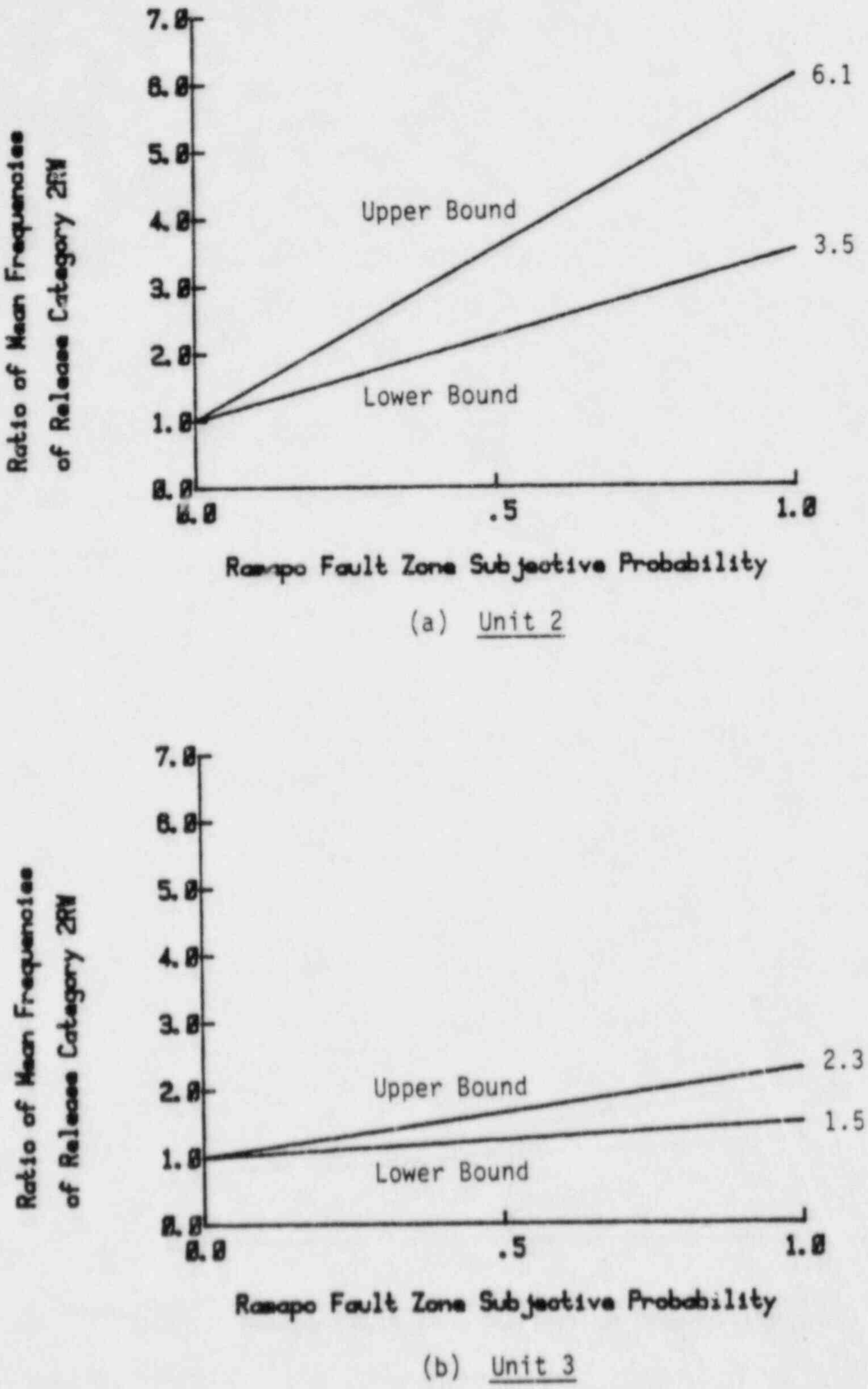


Figure 7 Effect of Including a Ramapo Fault Zone on Total Release Category 2RW

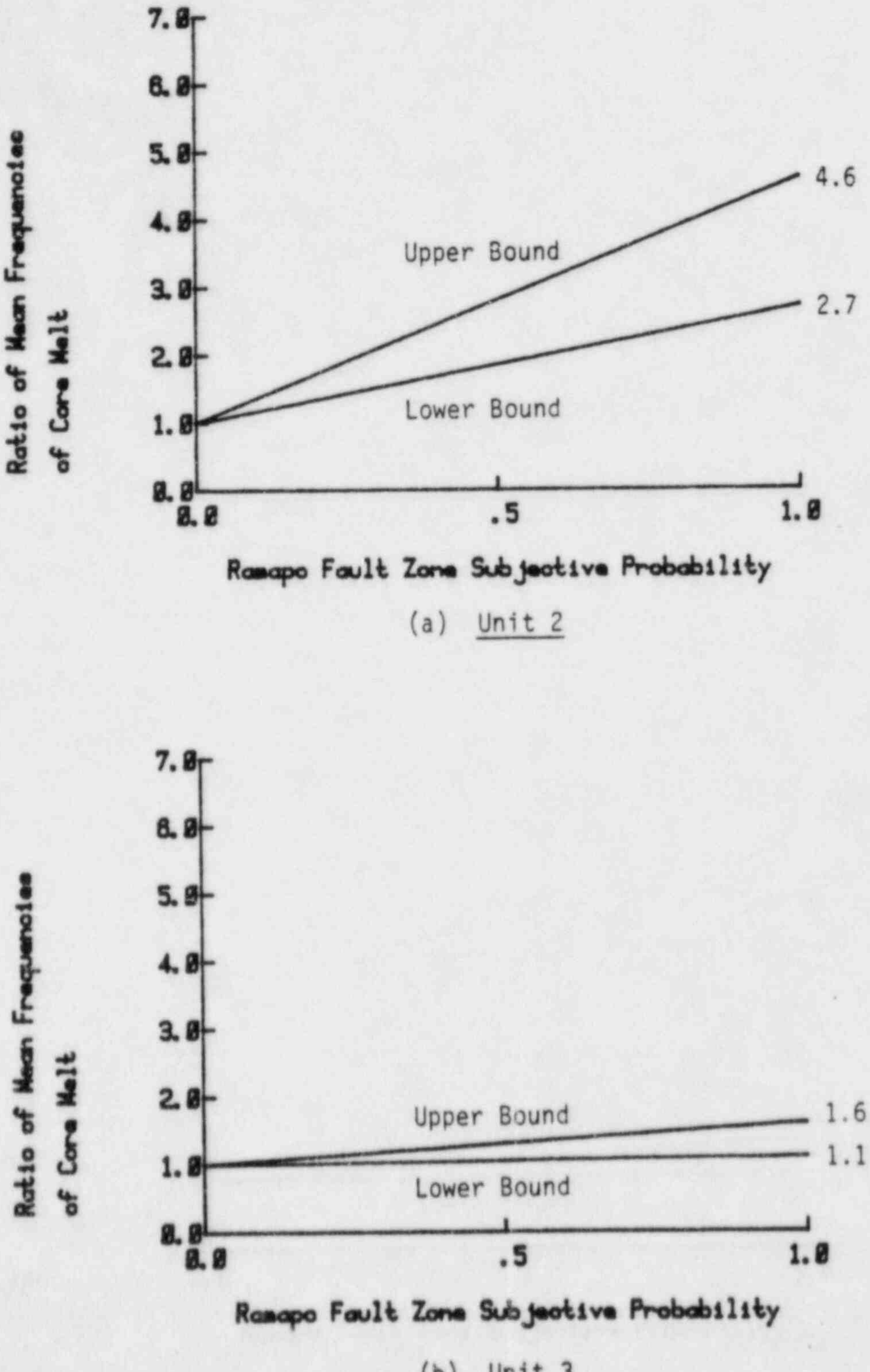


Figure 8 Effect of Including a Ramapo Fault Zone on Total Core Melt

5. CONCLUSIONS AND RECOMMENDATIONS

Based on our review of the IPPSS, we believe that certain results may be unconservative. This chapter gives conclusions concerning the frequency of core melt and the various release categories. We also give recommendations to resolve the most significant issues which we have raised in the review.

Table 1 gives a revised list of mean frequencies for Indian Point Unit 2 based on our review. Table 2 gives a similar list for Unit 3. Below each of the mean frequencies for seismic, hurricane, and tornado is the ratio of the revised value to the value given in the IPPSS report (see Tables 8.3-2 and 8.3-3 for the IPPSS report values for Units 2 and 3, respectively). Flooding is not included in either table. We believe that detailed probabilistic analyses should be conducted for both internal and external flooding. At this point, we do not have sufficient information to quantify the effects of potential flooding.

The following sections summarize our basis for the revised frequency values given in Tables 1 and 2.

INDIAN POINT UNIT 2

The basis for the revised frequency values for seismic, hurricane, and tornado for Indian Point Unit 2 are given below.

Seismic

We believe that the D&M hazard curve values are more representative of the Indian Point site; thus, we choose to weigh the D&M curves with a probability value equal to 1.0. This assumes that the results for a Ramapo source are contained within the family of seismicity curves developed by D&M. Because the release category frequency values for the WCC curves are about an order of magnitude below the frequency values based on using the D&M curves, the mean release category and core melt frequencies are approximately doubled. (Remember that the D&M and the WCC values were each weighted equally in the IPPSS.)

Hurricane

Two factors produce an estimated increase in release category 2RW. Due to a higher estimated hazard curve, the frequency of 2RW and core melt are judged to increase by a factor of 10. Also because offsite power probably will be lost at a wind speeds below 140 mph, the frequency of 2RW release and core melt increase by a factor of 2. The total factor for both these effects is a 20-fold increase in mean frequency for 2RW and core melt.

Tornado

Similar to the hurricane analysis, we believe that the capacity of offsite power has been assumed too high. We estimate that the frequency of release category 2RW increases by a factor 2. However, we judge that the hazard curves are conservative by at least an equivalent factor; thus, we believe that the IPPSS mean frequency values for 2RW and core melt are reasonable.

Summary for Unit 2

The total mean frequencies (given in the last column of Table 1) revised for seismic, hurricane and tornado effects for release categories Z-1Q, Z-1 and 2 increase by a factor of 2 or less. For release category 2RW the increase is estimated to be a factor of 3. The total effect on core melt is a factor of 2.3 increase. These changes are judged to be small to moderate.

INDIAN POINT UNIT 3

The basis for the revised frequency values for seismic, and discussion for hurricane and tornado effects for Indian Point Unit 3 are given below.

Seismic

We believe that the capacity of the hung ceiling in the control room may be lower than the equivalent median capacity value of 0.8g, implicitly used in the IPPSS. We estimate that the mean frequency for release category 2RW, which has a dominant contribution from the control building, increases by a factor of 5. Similar to the revised values for Unit 2 for the increase in the hazard function, we increase the mean frequencies of all categories by an additional factor of 2 to produce a total factor equal to 10 for release category 2RW and a factor of 2 for other categories. Core melt due to seismic increases by a factor of almost 8.

Hurricane

This is not a significant event for Unit 3.

Tornado

Since the frequency of release depends on tornado missile impact, we judge the IPPSS results to be reasonable.

Summary for Unit 3

As shown in Table 2, the revised release category total mean frequencies for the effects of seismic are changed at most by a factor of 2. The total effect on core melt is only a factor of 1.2 increase. We consider the revised mean frequency values for Unit 3 to be small.

RECOMMENDATIONS

In order to resolve the most significant issues which have been raised in the review, we recommend the following be done.

Seismic

1. For Unit 3, the capacity of the hung ceiling in the control room should be analyzed and a fragility curve developed for this component and incorporated into the plant analysis.

2. For Unit 3, the capacity for the diesel generator fuel oil tank, which is a dominant contributor, should be based on a specific analysis for this component. Generic-based values were used in the IPPSS.

3. The Ramapo Fault should be represented in the seismic hazard analysis (i.e., area, recurrence distribution, upper-bound magnitude, etc.) and probability weight(s) assigned.

Flooding

1. A probabilistic analysis should be conducted to consider the variability in important parameters of the flood process that determine the flood profile, and which also takes into account the uncertainty in the frequency of flooding.

2. A more detailed and systematic presentation of the method used to evaluate the impact of internal flooding should be included in the IPPSS. This presentation and the results of the analysis should be integrated with the plant fault trees such that the impact of flooding is clearly represented and accounted for in the analysis.

Wind

1. A fragility curve for offsite power should be developed which considers various possible failure mechanisms (i.e., in addition to the failure of the transmission towers).

2. Wind fragility curves should be rationally developed for the Unit 2 control building and the diesel generator building. They should explicitly consider the structure shapes and the effects of adjacent structures. Possible local failure of siding and roofing should be considered in determining the structure capacities. Also, the fragility of the Unit 1 turbine and superheater buildings should be calculated for wind. The possibility of these buildings failing and falling on safety-related structures (i.e., Unit 2 control building, diesel generator building, and condensate storage tank) should be included in the plant analysis.

3. A hurricane hazard analysis which includes careful evaluation of the site roughness boundary layer effects and wind channelization by the local hills and river valley should be performed.

TABLE 1.

REVISED MEAN RELEASE FREQUENCIES - UNIT 2

Mean Frequency
(Ratio to IPPSS Report Value)

Release Category	Seismic	Hurricane	Tornado	Other*	Total
Z-1Q	1.4×10^{-6} (2)	0	0	0	1.4×10^{-6} (2)
Z-1	2.6×10^{-8} (2)	small	small	3.5×10^{-9}	3.0×10^{-8} (1.8)
2	5.8×10^{-8} (2)	small	small	5.1×10^{-7}	5.7×10^{-7} (1.1)
2RW	2.8×10^{-4} (2)	5.4×10^{-4} (20)	1.6×10^{-5} (1)	1.4×10^{-4}	1.0×10^{-3} (3)
8A	8.4×10^{-9} (2)	0	small	4.8×10^{-5}	4.8×10^{-5} (1)
8B	5.2×10^{-10} (2)	0	0	9.1×10^{-5}	9.1×10^{-5} (1)
Core Melt	2.8×10^{-4} (2)	5.4×10^{-4} (20)	1.6×10^{-5} (1)	2.9×10^{-4}	1.1×10^{-3} (2.3)

*Includes Fire and Internal Events (IPPSS Table 8.3-2, pg 8.3-14)

TABLE 2.
REVISED MEAN RELEASE FREQUENCIES UNIT 3

Release Category	Mean Frequency (Ratio to IPPSS Report Value): Seismic	Hurricane	Tornado	Other*	Total
Z-1Q	7.4×10^{-8} (2)	0	0	0	7.4×10^{-8} (2)
Z-1	5.0×10^{-9} (2)	0	small	2.3×10^{-9}	7.3×10^{-9} (1.6)
2	small	0	small	4.9×10^{-7}	4.9×10^{-7} (1)
2RW	2.4×10^{-5} (10)	0	9.2×10^{-7} (1)	6.2×10^{-5}	8.7×10^{-5} (1.3)
8A	1.4×10^{-6} (2)	0	4.1×10^{-7} (1)	5.8×10^{-9}	1.8×10^{-6} (1.6)
8B	4.4×10^{-7} (2)	0	0	1.3×10^{-4}	1.3×10^{-4} (1)
Core Melt	2.6×10^{-5} (7.9)	0	1.3×10^{-6} (1)	1.9×10^{-4}	2.2×10^{-4} (1.2)

*Includes Fire and Internal Events (IPPSS Table 8.3-3, pg 8.3-15)

APPENDIX A

REVIEW OF RONALD L. STREET

REVIEW OF THE INDIAN POINT SEISMIC HAZARD STUDY

prepared by

R. Street

July 09, 1982

TABLE OF CONTENTS

PART I

EVALUATION OF OVERALL METHODOLOGY

Evaluation of Overall Methodology

The methodology used in determining the frequency of the various levels of ground motions that might be experienced at the site is, in my opinion, fairly comprehensive. There is, however, one omission in the study that I feel needs to be addressed. That omission is the failure to do a background study of the more significant earthquakes within the general region of the site; i.e., the epicentral intensity VI and VII events in southeastern New York and northeastern New Jersey.

Such a study would have (1) suggested the possibility of low magnitude, high intensity, shallow events, (2) the likelihood that the August 10, 1884 earthquake is a 5 1/2 to 5 3/4 m_{bLg} event, and (3) would have corrected any errors with respect to the published reports about the earlier events.

With respect specifically to the many seismogenic zones considered in this study, my personel preferences are the ones labeled Sounce Area 1 and Source Area 5 in the Woodward-Clyde report. As for the proposed Ramapo fault zone, and after reviewing the paper by Aggarwal and Sykes (1978) and the larger events that they suggest might be associated with that fault zone (see Table IV-1), it is my opinion that the proposed zone is speculative and should according be assigned a low level of probability (i.e., <0.2).

Reference

Aggarwal, Y. P. and L. R. Sykes (1978). "Earthquakes, faults, and nuclear power plants in southern New York and northern New Jersey", Science, 200, 425-429.

PART II

INFLUENCE OF FINDINGS ON FINAL RESULTS

Influence of Findings on Final Results

In Part IV, Section B, the results of a cursory review of six events within the general region of the site are tabulated. One important result with respect to this study is the 5 1/2 -5 3/4 m_{bLg} magnitude estimated for the August 10, 1884 event. Depending upon where one chooses to place the epicenter of the event, the maximum magnitude event in the Northeast Tectonic Zones listed in Table 1 of the Dames and Moore report would need to be raised. Given the difficulty to make a definitive statement about the epicentral location of a non-instrumental event, I recommend that the mean $m_{b,\ max}$ of the Highlands and Conestoga Valley Tectonic Zones be raised to 5.7.

With respect to the low-magnitude, high intensity earthquakes that are noted elsewhere in this review, both the Dames and Moore ($I_o \rightarrow m_b \rightarrow a$) and Woodward-Clyde ($I_o \rightarrow I_s \rightarrow a$) technique for estimating ground motion, tends to overestimate the motion in the far-field. However, there does remain a question in my mind, as to what is the character of the ground motion of a 3.0 magnitude event that can cause intensity VI MM effects (event 06 October, 1971 in Table 1 of the Woodward-Clyde report) in the epicentral region.

PART III

EVALUATION AND COMMENTS ON THE

DAMES & MOORE AND WOODWARD-CLYDE REPORTS

Evaluation and Comments on the Dames & Moore and Woodward-Clyde Reports

I have reviewed both the Dames & Moore and Woodward-Clyde reports with respect to the techniques they used to estimate the acceleration as a function of earthquake magnitude/epicentral intensity and distance in detail. And while the two approaches differ, both reports rely heavily upon current and generally accepted scaling techniques.

The conclusions of both studies, however, are based on the acceptance of the single parameter -epicentral intensity- to characterize the regional seismic activity. It is with this portion of both studies that I disagree. As discussed in Part IV, clearly there are earthquakes of appreciably different magnitudes and extent of areal damage, but which are considered equal if judged on the basis of their epicentral intensities. It is my opinion that this approach too simplistic, and that the report would be greatly strengthened if the more significant events in the area were defined in greater detail.

PART IV

COMMENTS ON THE REGIONAL SEISMICITY

Comments on the Regional Seismicity

Due to the lack of instrumental data, the frequency of the various levels of ground motions at the site considered in this study were derived by converting the catalogued epicentral intensities to $m_{b(Lg)}$ magnitudes, or by the derivation of a site intensity by means of a spatial attenuation-of-intensity relationship. Both approaches make the assumption that the regional seismicity can be characterized by a single parameter, the maximum epicentral intensity. Yet, there are several instances in both the historical and instrumental data base when earthquakes of appreciably different magnitudes and extent of areal damage are catalogued as having the same epicentral intensity; i.e., the August 10, 1884 and the June 01, 1927 events described in Table IV-1. Yet, by the methodology used in this study, such events are considered with equal weight.

In my opinion, therefore, a weakness in the seismicity section of the seismic risk analysis done for this study is the lack of a detailed review of the more significant events in the region. By detailed review, it is meant the documentation of the effects of the earthquakes via published reports, newspapers, etc., and the use of instrumental records where available. If a definitive study had been done on the more significant events, it is quite likely that several of the assumptions utilized in this study could have been tested against the observations. For example, a definitive study of the August 10, 1884 event would have resulted in data base by which to test the spatial attenuation of intensity relationships referred to in the study. A definitive study of the 1884 event, would also most likely have suggested an earthquake in the range of a 5 1/2 to 5 3/4 m_{bLg} magnitude event, rather than the 5 1/4 m_{bLg} magnitude estimated by the relationship:

$$m_b = 0.5 (I_o + 3.5)$$

A review of the seismograms of the March 23, 1957 event, interpreted in accordance with the m_{bLg} magnitude formula developed by Nuttli (1973A), would have indicated a magnitude more on the order of a 3.3 event, rather than the 4 3/4 m_b derived by the above formula. And a review of the intensity data published in the 1957 edition of United States Earthquakes, would have demonstrated the inadequacy of the attenuation of intensity relationship used in this study.

Table IV-1 gives the results of a cursory review of six of the more significant events in the general region as part of a background study I undertook as part of this review. The results listed in this table are not meant to be definitive, which is beyond the scope of this review, but rather as an indication of the type of information that is available and which in my opinion should have been incorporated in the study.

TABLE IV-1

RESULTS OF A CURSORY REVIEWS OF

THE EARTHQUAKES OF:

December 19, 1737
November 30, 1783
August 10, 1884
September 01, 1895
June 01, 1927
March 23, 1957

December 19, 1737

Based on the description of this event in Coffman and von Hake (1973) as being felt from Boston, MA to New Castle, DE, it is estimated that the felt radius of this event was on the order of 260 km. Using this radius to estimate the felt area (this includes a hypothetical offshore area), and Formula (5) in Street and Lacroix (1979), the m_{bLg} magnitude for this event is calculated to be 4.8 ($\pm$0.30).

November 30, 1783

The m_{bLg} magnitude estimated for this event was obtained by comparing the newspaper reports for this event at Boston (MA), Hartford (CT), New Brunswick (NJ), New Haven (CT), New London (CT), Philadelphia (PA), and Worester (MA), to the newspaper reports for the August 10, 1884 earthquake. In general, the newspaper reports indicated that the 1783 event was experienced at about one intensity unit less that experienced during the 1884 event. The m_{bLg} was then calculated by adjusting the falloff-of-intensity curve obtained for the 1884 event downwards one intensity unit. This resulted in a m_{bLg} magnitude of 5.2

Sources of Information:

Boston Gazette
Connecticut Courant
Connecticut Journal
New London Gazette
Pennsylvania Packet

August 10, 1884

The m_{bLg} magnitude of the August 10, 1884 event was estimated on the basis of the felt area, the area within the intensity IV isoseism, and the falloff-of-intensity technique developed by Nuttli (1973B). Based on a felt area estimate of 557,000 sq km (includes a hypothetical offshore area), a 187,000 sq km area within the intensity IV isoseism, and a falloff-of-intensity m_b estimate of 5.7, the m_{bLg} magnitude of the 1884 event is estimated to be 5.6 ($\pm$0.15) on the basis of Formula (7) of Street and Lacroix (1979).

Sources of Information:

Rockwood (1885)
Albany Evening Journal
Albany Evening Union
The Albany Times
Amsterdam Daily Democrat
The Baltimore Morning Sun
Elmire Weekly Advertiser
The Globe and Mail
New York Herald
New York Times
The Plattsburg Sentinel
Rochester Democrat and Chronicle
Rochester Union and Advertiser
Rome Daily Sentinel
Sunday Morning Tidings
Troy Times
Utica Morning Herald
(and Daily Gazette)
Washington Post

September 01, 1895

The m_{bLg} magnitude for the September 01, 1895 event is estimated on the basis of a 45,000 sq. km felt area, which by Formula (5) of Street and Lacroix (1979) yields a 4.3 m_{bLg} magnitude.

A problem with this event, are the conflicting reports as to the extent of the felt area. Coffman and von Hake (1973) indicate that the event was felt from Virginia to Maine, but newspaper accounts indicate that the event was felt no further north than Sing Sing, New York, and no further south than Wilmington, DE.

In this review, I have chosen to use the newspaper accounts as a basis for calculating the magnitude.

Sources of Information:

The Baltimore Morning Sun
New York Herald
New York Times

June 01, 1927

The 3.9 m_{bLg} magnitude for the June 01, 1927 event is estimated on the basis of 16,000 sq. km felt area (includes a hypothetical offshore felt area) and Formula (5) of Street and Lacroix (1979).

A 4.2 upper bound on the magnitude can be estimated on the basis of the fact that the earthquake was not recorded on the Galitzin (Cambridge Type) seismometers located at Georgetown University, $\sim$300 km to the south. The magnification of the vertical Galitzin is known to have been $\sim$360 at 1 Hz.

Sources of Information:

New York Times
Washington Post

March 23, 1957
OT(UT): 19-02-31
40 3/4°N/74 3/4°W

STATION	TYPE OF INSTRUMENT	EPICENTRAL DISTANCE (km)	AMPLITUDE (mm)	PERIOD (sec)	MAGNIFICATION	m_{bLg}
WES	BENIOFF	338	1.4	0.6	30K	3.1
OTT	BENIOFF	521	2.5	0.4	65K	3.4*
MNT	BENIOFF	535	2.5	0.3	63K	3.6*
SFA	WILLMORE	779	0.5	0.7	27K	3.1

Felt area is estimated to be ∿ 2,000 km^2. Using Formula (5) in Street and Lacroix (1979), felt area suggests a m_{bLg} magnitude of 3.4

Average m_{bLg} = 3.3

*Note: the 3.4 and 3.6 m_{bLg} magnitudes from the records at OTT and MNT are probably inflated due to the use of the 0.4 and 0.3 second periods.

**Felt Area based on information in United State Earthquakes (1957).

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Coffman, J. L., and C. A. von Hake, Editors (1973). Earthquakes History of the United States (revised edition through 1970), Publication 41-1, Environmental Data Service, NOAA, U.S. Dept. of Commerce, Boulder, Colorado, 208 p.

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APPENDIX B

REVIEW OF ERIK H. VANMARCKE

CRITICAL REVIEW

of the

INDIAN POINT PROBABILISTIC SAFETY STUDY

(SEISMICITY AND SEISMIC RISK)

by

Erik H. Vanmarcke

Submitted to

J. R. Benjamin & Associates

Mountain View, California

July 9, 1982

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EVALUATION OF OVERALL METHODOLOGY

The overall analysis format involving consecutive matrix operations on the vector(s) of initiating event probabilities is simple and attractive, and it is quite appropriate for seismic risk evaluation.

Seismicity Studies

In the seismicity study, the basic format of generating a family of seismicity curves to which subjective weights are assigned is sound, and the assignments of equal weights to the two studies (Dames & Moore and Woodward-Clyde) is reasonable. The two studies are based on similar methodology and yield quite comparable site seismicity estimates.

The seismicity analysis output is expressed in terms of a simple ground motion parameter (acceleration). In this format, other ground motion parameters which may have a significant effect on system response and performance are ignored: strong-motion duration, parameters of the frequency content (e.g., dominant frequency or the ratio of peak velocity to peak acceleration). Ideally, the output of a site seismicity analysis would be the multi-dimensional distribution of a vector of ground motion parameters.

The Cornell (1968) seismic hazard model used in the Indian Point study integrates statistical and tectonic information about earthquake occurrence. Contrasted to an approach based directly on historical epicentral locations and magnitudes, the method permits (in fact, necessitates) an expression of judgment about the location and geometry of seismogenic zones (zones where earthquake occurrence is believed to be of similar tectonic origin).

The method is most potent in regions where tectonic evidence is strong, e.g., where the presence of sources (usually faults) is undisputable. In the Northeastern U.S., where there is much controversy about the causes and mechanisms of earthquakes, there is great diversity of (relatively soft) expert opinion about seismic zonation. Therefore, when the seismic risk results based on the different interpretations of the historical seismicity (i.e., different assumptions about seismic zonations) are eventually weighed, the composite seismic risk is bound to be close to that obtained from an approach based purely on historical seismicity.

The process of selecting seismogenic zones is tantamount to assigning a specific spatial distribution to the historical seismicity in the area surrounding the site. When a new seismogenic zone is introduced, the mean seismicity activity (number of quakes per year) assigned to it will be taken away from the other seismic zones already included in the model, as the aggregate seismic activity (the sum over all zones) tends to remain close to the historical value.

It is in this light that one can examine the question as to whether the Ramapo Fault (described by Aggarwal and Sykes, 1978) should have been included as a seismogenic zone. If one sees as the main effect introducing a new source, the diversion of some of the seismicity away from the other seismogenic zones, no significant differences in predicted site seismic risks should be expected.

For the same reason, it is reasonable to ignore the uncertainty about the activity rates of individual sources. Individual source contributions to site seismic risk are approximately linearly dependent on the source activity rate. Hence, the effect of varying source activity is easy to assess, and the variability of this parameter does not have much impact on overall uncertainty in site seismicity.

A variety of attenuation relationships are considered which reasonably represent available information about site motion intensity in function of magnitude and distance. The lognormal distribution is the standard model for the "error factor," and the assumed value for the log-standard deviation (σ = 0.6) is also reasonable.

The Indian Point plant is founded on bedrock. This fact is not accounted for in the Dames & Moore study. In the Woodward-Clyde analysis, the preferred attenuation relationship (proposed by Cornell and Merz) has the advantage of being specifically applicable to ground motion on bedrock.

The most controversial aspect of the seismicity study is the imposition of an upper bound on effective ground acceleration. Although there is merit to the arguments advanced in the report (Section 7.9.4) by Structural Mechanics Associates, Inc. (which form the basis for the upper bounding of effective acceleration), they really constitute one expert's opinion. Other experts would likely disagree with the proposition that such bounds exist. In effect, in what is somewhat a matter of professional judgment, the full weight is assigned to a single expert opinion (expressed in Section 7.9.4).

The relative frequency of occurrence of earthquakes is represented by a truncated exponential distribution. A better model might be one in which the frequency-versus-magnitude law decays more gradually with magnitude. This would certainly lessen the impact of the imposition of an upper bound magnitude (Dames & Moore) or an upper bound Modified Mercally Intensity (Woodward-Clyde).

The key question is not whether the Ramapo fault is included or not, but if there are perhaps reasons to assign different (unfavorable) values to its seismicity parameters, in particular, the upper bound magnitude $M_{b,max}$. It is doubtful that there is evidence that would support unusual values for the seismicity parameters of the Ramapo fault.

Treatment of Fragility in Seismic Safety Analysis

In reference to the five main steps in the seismic safety analysis (as outlined in Section 7.2.1), I would argue that there is a missing step. In between Step 1 (Seismicity) and Step 2 (Fragility), there should be a step labeled "Seismic Response" or "Seismic Load Effect."

When an earthquake occurs, a ground motion characterized by peak acceleration a (whether "instrumental," "effective," or "sustained" does not matter at this point) is experienced at the base of the structure. The dynamic seismic input causes many simultaneous response accelerations a_j at points j (locations of structural components or equipment support points) throughout the structure. These response motions have frequency content quite different from that of the input motion. The output-to-input acceleration ratios a_j/a may be seen as random variables whose marginal statistics depend on the seismic response, the randomness of the ground motion, the (uncertain) dynamic properties, etc. Seismic design is based on the predicted response accelerations a_j to which an appropriate safety factor is applied. This yields the mean or median capacity (or resistance) of component j expressed in terms of acceleration. The actual capacity of component j is of course a random variable.

In the format of the seismic safety part of the Indian Point study, the uncertainty represented by the fragility curves originates from both the loading and the resistance, and the uncertainty about the (response related) ratio a_j/a is incorporated in the fragility curves. The introduction of an intermediate step (Seismic Response or Seismic Load Effect) in the seismic safety assessment would help clarify and resolve many issues related to modeling, interpretation and processing of component fragility curves, in particular:

(a) Variability: The components of uncertainty related to seismic input (owing to complexity of accelerograms) and response could be separated from those related to capacity or resistance (measurable by component testing).

(b) Probability Models: Much is known about probability density functions of seismic load effects. It would no longer be necessary to adopt the sweeping assumption that all random variables involved have a lognormal distribution.

(c) Failure Criteria: It would become unnecessary to express all fragility curves in terms of peak acceleration (a definite drawback of the present format). Depending on the function (or rather, malfunction) of each component, the fragility curve might be in terms of maximum (response) acceleration, sustained peak acceleration, relative displacement, or even energy absorption capacity.

(d) Correlation: Patterns of correlation (different for random load and resistance factors) are not adequately accounted for in the present format of converting component fragility curves into system fragility curves by using plant logic diagrams. The component-to-system conversion is now accomplished (quite artificially) in the "resistance domain" by assuming statistical independence between the random variables that control the width of component fragility curves. In reality, for a given input acceleration, the response accelerations a_j are fairly strongly correlated, while the associated component resistances are perhaps more nearly independent. Depending on the relative variability of load effects and resistances, the real system condition would be closer to one or another of the two extreme conditions of perfect dependence and perfect independence.

DETAILED REVIEW COMMENTS

SECTION 7.2.2 (Seismicity)

Most of the detailed review comments about the seismicity study are presented as part of the review of Sections 7.9.1 and 7.9.2. My main concern with the summary in Section 7.2.2 is that it does not clearly indicate how the final family of seismicity curves was obtained. Nowhere in the Dames & Moore report are rigid bounds imposed on effective peak acceleration; this asymptotic behavior at low risk levels is, however, the single most striking feature of seismicity curves in Figure 7.2-4. The last paragraph in Section 7.2.2.1 does not adequately explain the logic which led to the final seismicity curves.

The terminology used to refer to the various measures of acceleration (peak, sustained, sustained-based and effective) is quite confusing. Note, for example, the final three paragraphs in Section 7.2.2.1. The presence of these different acceleration measures and correction factors points to the urgent need to implement improved earthquake ground motion descriptions that explicitly account for duration (in addition to a measure of intensity such as peak acceleration) and to apply analysis procedures which predict seismic response measures more directly correlated with performance and damage. Much of this is within the state-of-knowledge of earthquake engineering.

SECTION 7.9.1 (Dames & Moore Seismicity Study)

Page 2, Seismic Hazard Model, Item 1: I question the statement: "The average predicted rates of occurrence in these zones are accurately estimated by historical occurrence in these zones." The words "predicted" and "accurately" should be dropped.

The comment (on page 5 end of 2nd paragraph) "even if peak accelerations are high" is revealing. It implies recognition that accelerations are indeed highly variable. Many seismologists and earthquake engineers would say that this is equally true at high as well as at low values of m_b (or Mercalli Intensity), and that any rigid upper bound on peak acceleration is unrealistic.

Uncertainty about the "b-value" for Northeast tectonic zones and for the Piedmont zone (on page 6): The three-valued discretization (mean and mean $\pm$ one standard deviation) may be inadequate as it obviously does not cover the tails of the distribution. No variation is assumed for the b-value associated with the Piedmont-Cape Ann zone.

Discretization of $m_{b,max}$ (on page 6): The double-triangular distribution has an upper bound of 6.2; it is then converted into a three-valued probability mass function whose largest value is $m_{b,max} = 6$. If the rigid bound on effective acceleration were to be relaxed, the resulting error in seismic risk calculations may not be negligible in the very low probability range.

It is stated on page 7 (paragraph 2) that "It was felt that there is some negative correlation between b-values and values of $m_{b,max}$." This is the justification for assuming complete probabilistic dependence between b and $m_{b,max}$. It would be interesting to see some results based on the assumption that b and $m_{b,max}$ vary independently. Also, it might have been preferable to quantify the seismological consultant's judgment in terms of a (discretized) joint probability distribution implying partial correlation.

Treatment of Peak Acceleration (page 8): Nuttli's data indicate that the 1.37 value for the ratio of sustained to peak acceleration applies to the magnitude range $m_b \leq 6.0$. The 1.37 value is in fact adopted for all magnitudes. Note, however, that the upper magnitude bound adopted in the study equals $m_{b,max} = 6.0$ (with probability 0.28), while the m_b magnitude follows a truncated exponential distribution; it follows that the condition $m_b \leq 6.0$ (to which the 1.37 value corresponds) is, in fact, assigned zero probability of occurrence. The 1.37 value is therefore subject to question.

The influence of the choice of a_{max} is understated, for example on page 12 in Section 7.9.1: "The variation in hazard resulting from the use of alternate estimates of peak acceleration is generally within the variation resulting from different b-values and $m_{b,max}$ values and from hypotheses on seismogenic zones." It is quite obvious from the final seismicity curves that calculated probabilities are more sensitive to a_{max} than to these other assumptions in the critical "high acceleration-low probability" range of the seismicity curves.

The assignment of uncertainty to the attenuation laws ($\sigma_{\ln a}$ = 0.6) is reasonable. Alternate assumptions could have been tested (with appropriate weights attached), but I expect this would not have had much impact on the final results. The same may be said about the choice of the lower limit on magnitude (m_b = 4).

SECTION 7.9.2 (Woodward-Clyde Seismicity Study)

The Woodward-Clyde study carefully considers a range of choices for the models and equations to describe source location and geometry, activity rates, upper bound magnitude (epicentral intensity) and attenuation laws. Preferred choices are identified in each instance (except for upper bound intensity), leading to a "base case" site seismicity output. The latter constitutes the recommended input into the plant seismic safety analysis.

The sensitivity analysis is limited to single changes in each one of the assumptions made in "base case." Although there is a very common way of doing sensitivity analysis, it is obviously limited and oversimplified. The different seismicity parameters of each seismogenic zone (e.g., size, activity, b-value upper bound magnitude) are strongly interdependent. Hence, varying one parameter (or making it more variable) in principle necessitates re-examination of all related parameters.

It is stated on page 8: "The composite value is not an accurate representation of our uncertainty regarding upper bound." Should it be is?

The most critical parameter is the upper bound intensity, selected to be either VII or VIII with likelihoods of 80 percent and 20 percent, respectively. This composite bound is used in the "base case" seismicity analysis. It would be nice to see results based on different sets of intensities and associated likelihoods. In particular, small weights could be assigned to intensities VI and IX.

The Cornell-Merz attenuation relationship has the advantage of being applicable to the Northeastern U.S. as well as to rock sites. This is consistent with the location and site condition of the Indian Point plant.

Ground motion attenuation is evaluated in two steps. Site intensity is first predicted as a function of epicentral intensity and distance, and is then converted to peak ground acceleration. Bounds on peak acceleration are introduced (forthrightly and explicitly, as an integral part of the seismicity analysis) in the second step. Nuttli's sustained acceleration is adopted as an appropriate representation of "effective peak acceleration." The report offers a thoughtful discussion of the rationale behind this choice.

The very crude discretization of acceleration (see Table on page 21) is questionable. It would have been preferable to consider an array of at least four or five accelerations for each intensity.

SECTION 7.9.4 (Structural Mechanics Associates, Inc., Damage-Effective Ground Accelerations)

While I agree with SMA's assessment of the inadequacy of peak acceleration to represent damage or damage potential (because factors such as ground motion duration and inelastic behavior are unaccounted for), I feel that the proposed acceleration reduction factors, and especially the upper bounds on acceleration, are introduced incorrectly at the end of the seismicity analysis. Such bounds (with probabilities attached) should perhaps appear in the attenuation laws, as part of the input to the seismic hazard analysis. As it is, they appear as an after-the-fact adjustment of the output.

SECTION 7.2.3 Fragility

The choice of the lognormal distribution is expedient but not necessarily consistent with available information. Seismic response itself is more nearly normal than lognormal. [Seismic excitations are approximately normal (with mean zero), and any linear system preserves this normality; hence, the response time histories are normal.] The absolute maximum of the random response of a linear system follows an extreme value distribution about which much is known. Hence, the sweeping assumption of lognormality is justified mainly on account of analytical convenience (i.e., it facilitates analysis of products of independent random variables).

Section 7.2.3.1 on "Definition of Failure" makes it clear that acceleration is not necessarily the best response parameter in terms of which to define fragility curves; for example, relative displacement or energy absorption capability may be preferable in some cases.

The evaluation of fragility is in many cases judgmental. For the critical components, it would be desirable to validate judgment through appropriate (nonlinear) dynamic analyses using as input time histories of ground (floor) motion.

EVALUATION OF FINAL RESULTS

In the detailed comments, I have tried to identify all the main assumptions made in the seismicity portion of the Indian Point PRA. Whenever possible, I have expressed my judgment about the appropriateness of each assumption and about its likely impact on the final results.

Both seismicity studies adopt the "seismogenic zone" approach rather than alternative methodology based purely on historical seismicity. In view of the range of assumptions about source locations and shapes represented in the two studies, I judge that the range of probabilities adequately covers what would be predicted by alternate methodologies. Taken together, the two seismicity studies produce a representative family of seismicity curves.

A reasonable set of weights has been assigned to the different seismicity curves (13 from the Dames & Moore study and 4 from the Woodward-Clyde study). The better treatment of ground motion attenuation and the more logical introduction of upper bounds on acceleration offset the more limited sensitivity analysis in the Woodward-Clyde study, and justify the assignment of equal weights to the two studies.

Overall, I believe that the results expressed in terms of mean annual risk of core damage (or mean risk to public health and safety) are quite insensitive to reasonable variations in the assumptions about seismic zonation, seismicity parameters and discretization intervals. While these assumptions are unlikely to have much impact on mean risk rates, they will affect (in ways hard to predict) the shape and the spread of the final "frequency-of-probability" curves.

The most critical assumption is that there is an upper bound on effective peak acceleration. Such bounds are seldom encountered in conventional seismic risk work. If this asumption were to be relaxed it will probably lead to moderate increases in final mean seismic risk estimates, and to a broader spread in the high acceleration end of the family of seismicity curves.

REFERENCES AND SOURCES USED

Sources

My report is based on my general experience in structural safety and earthquake engineering. Among the documents on which I relied are the technical reports of an NSF project on Evaluation of Seismic Safety of Buildings which ran from 1974 to 1978 and for which I served as principal investigator.

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APPENDIX C

REVIEW OF LARRY R. RUSSELL

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REVIEW OF INDIAN POINT PRA, CHAPTER 4 AND APPENDIX C

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12 July 1982

SUMMARY AND CONCLUSIONS

The median and upper bound hurricane wind probabilities estimated for the Indian Point location in the PRA appear to be low for the rarer events (recurrence intervals greater than about 200 years). This apparent underestimation arises from treating all wind source directions in the same way, without allowing for more severe conditions from certain directions. Better estimations of these probabilities will require careful evaluation of complex site roughness boundary layer effects and wind channelization by the local hills and river valley. By coupling appropriate wind adjustments by direction with the generally satisfactory existing model, more accurate wind estimates can be produced. It is recommended that such a reanalysis be made to account for these effects.

DISCUSSION

ADEQUACY OF APPROACH

The basic simulation approach utilized in the PRA is an adaptation of the standard method utilized to estimate hurricane wind probabilities in regions of insufficient data. The PRA simulation approach is sufficiently accurate, with appropriate data and wind formulas, to estimate wind recurrence probabilities for critical structures, The input data selection for the PRA was realistic or somewhat conservative (i.e., tending to produce higher estimated than the raw data would yield.) The storm decay rate when moving inland and the storm occurrence rates were fairly conservative. These overestimates tend to offset any underestimation errors in the windfield estimates due to lack of allowances for the local wind intensificaitons from rainbands.

The primary determinants of the wind, given a hurricane occurrence, are the central pressure drop of the storm, its size and forward speed, and the influence of the site topography and boundary layer roughness. The treatments of the central pressure drop, storm size and forward speed were accurate and conservative. The primary question of accuracy relates to the treatment of site conditions.

INFLUENCE OF LOCAL TOPOGRAPHY

The Indian Point Plant is in an unusual location where it is very difficult to evaluate site conditions without model studies. The broad Hudson River Valley to the southeast of the site, along with the adjacent hills, will tend to channel winds up the river. Such winds will be higher than those elsewhere in the area due to both reduced friction overwater and channelization effects. Winds from the southeast, southwest and south will tend to blow past the waterfront area from the south. Some lesser wind accentuation can also be expected from the northeast. For other wind directions, the wind values will likely be less or no more than would be expected in flat country. Field verification and quantification of the site friction and channelization effects is difficult due to lack of appropriate storm conditions. No appropriately sited weatherstations have been in the area long enough to provide good data for estimating these directional effects.

The general location of the site is in a rather sheltered region, where most storms travel parallel to or somewhat away from the coast. Storm proximity to the coast or passage inland for extended periods tends to appreciably weaken a storm. Between 1871 and 1980, only one storm (in 1903) has entered New Jersey or New York on a heading (west of north) where storm decay would be minimal and maximum storm winds would come overwater to the site. Most of the area storms which are not weakened by moving ashore on a northerly or northeasterly heading over New Jersey will move ashore over Long Island or farther east. In such cases, the site will experience either a weakened storm or will be on the weak side of an unweakened storm. When the site is on the weak side of a hurricane passing over Long Island, the site winds from the northeast will probably be increased over those predicted by the PRA model. If a north-to-northeasterly heading storm is relatively undecayed during its passage over New Jersey, then strong SE winds in excess of the PRA model preditions are also possible. These situations of either a relatively undecayed storm passing over New Jersey or a storm heading NW or NNW or N to the west of the site are rare, based on the historical data.

WIND PROBABILITY ESTIMATES FROM OTHER SOURCES

Other estimates of the winds at the location can be made by using NBS ldg. Science Series 124: "Hurricane Wind Speeds in the United States." Figure 1 indicates the windspeeds simulated for an open-country location 50 miles inland from the coast at New York City by the NBS. Given the general path trend for storms in the vicinity, the speeds simulated should be applied to sites to the NNE and NE of New York City. Somewhat reduced (by 10 to 15%) winds would typically be expected at the Indian Point site if it were in open country. It is felt by this reviewer that the open country wind estimates are representative for Indian Point site winds from the NE and the SE to SW quadrant. With adjustments for the proportion of storm wind maxima coming from other, more reduced wind directions, the resultant curve is still likely to exceed the PRA upper bound curve (PRA Figure IV-8) for the rarer events. When rare relatively undecayed strong storms pass over the length of New Jersey or a rare strong storm crosses the coast more directly and passes to the west of the site, the winds at the site will probably exceed 110 mph. While such storms are uncommon, they are likely enough to cause the actual hurricanewind risk curve to exceed that of the PRA upper bound. The median curve of the PRA appears considerably below the upper bound curve.

Emil Simiu summarized [A-1] observations taken from various sources in the New York-New Jersey area for hurricanes. New York City experienced 100 mph. winds in 1944, while Brookhaven on Long Island measured 95 mph. winds in 1954 and 115 mph. gusts in 1960. The elevations and exposures are not specified, but these coastal winds are still fairly high. More significantly, Trenton, New Jersey experienced 57 mph. from Hazel in 1957 and 56 mph. from Donna in 1960. The measurement location is in the city so these observations would correspond to about 70 mph. in open country. Trenton is about 50 miles inland and experiences essentially the same hurricane exposure, decay, and risk as the Indian Point site. The open exposure of the Indian Point site for NE and SE-SW winds could be expected to result in higher winds than for Trenton. The Trenton measurments tend to support the validity of the Batts, et al. [30] estimates for Indian Point. Use of NOAA Techinical Report NWS 23 [A-2] yields upper bound estimates for winds at Indian Point of about 146 mph. fastest mile from the SE. This estimate is based on a very extreme central pressure drop of 3.35 inch Hg which has never been approached in the New York area. The storm would move Northwest with the strongest portion of the windfield passing over the site. This highly improbable event is combined with more reasonable storm speed and size parameters and credible reduction of the site winds to 90% of those experienced at the coast. The "Standard Project Hurricane" (SPH) from the same reference would yield about 103 mph. fastest mile for a storm giving winds from an open country exposure at the site. The probability of a SPH is likely about 0.003/yr.

H.C.S. Thom prepared estimates of wind recurrences for several stations in the U.S. [A-3, A-4]. His estimates are based on observations at meteorological stations. Figure 2 shows his estimates for the Long Island seaward coast. These curves include phenomena other than hurricanes and serve as general indicators of conditions up to recurrence intervals of about 500 years. The values found by Thom should roughly approximate or exceed somewhat conditions at Indian Point for the severe directions if they are valid. However, Thom's results are known to be biased toward low values in some locations due to data censoring.

In general, it is felt that the estimates made in the study by Batts, et al. [30] provide a better estimate of Indian Point hurricane winds than do those of the PRA. The reason for this is that the site exposure is more nearly described by Reference [30]. Away from the river, it is felt that the PRA results are conservative.

CHAPTER IV REVIEW

A. Introduction: Adequate.

B. Methodology:

1. Hurricane Risk Model:
Equation (2) derived from Eqn. (1), is reasonable and conservative. The general approach of this section is reasonable and adequate. Equations (5), (6), (7) provide a satisfactory treatment.

2. Cyclonic Windfield Model:
The approach described in Equations (8) - (28) is acceptable for storms which are well behaved and smoothly varying, in the sense of not having any localized zones of relatively more intense convection. Any such localized zones of intensified convection will produce deviations from this windfield model.

Hurricanes are noted for their non-smooth variations, which occur in "rainbands". These bands are readily observed both visually and on radar as regions of heavier cloud density which have associated

2. Cyclonic Windfield Model (cont'd)

higher winds. While the peak winds of the storm may not be influenced much by such rainbands, the peak winds observed away from the radius of maximum winds likely will typically be higher than indicated by this model by 5-10 mph. Reference [A-2] indicates empirical results of windfield studies for hurricanes.

Development of Δp distributions directly from existing data is feasible, and probably more desirable, but use of a fixed p_n value is acceptable.

c. Use of an average value of K in Equation (28) is not reasonable. The values of K given in PRA References [30] and Kraft [35,40] are relatively high because the hurricane windfields used to estimate K have been found to justify such higher numbers. The study by Russell [26] uses an old windfield determination formula and should be excluded from the data set.

d. The formula for open country fastest 1-minute average winds used in Batts et al. [30] (with K=10.8) yields a higher wind estimate than that resulting from the average K=9.59. This higher value of K has been found to occasionally underestimate storm winds on the Texas coast. [A-5].

e. Decay: The rate of decay of the central pressure difference will be quite random in nature, but the source data sample is so limited that determination of an average value is difficult. The value used by Batts, et al. [30] was specifically chosen for a flood-prone flat coastal plain with very large bays. This value is unrealistically low for the Hudson Valley region, where large hills, lack of appreciable open water, and predominantly dry continental air sources would generally produce a faster than normal decay. The decay numbers used in the PRA will tend to underestimate the reduction of the hurricane strength as it moves inland. That is, the storm decay function tends to be conservative.

f. Maximum Windspeed During Storm Passage: The numerical procedure used to select a maximum is satisfactory.

3. Simulation Technique:

The simulation technique follows locigally from the analytical model selected. However, the development of the confidence limits for the simulation should not be taken as including all possible sources of bias or scatter. A more thorough discussion of possible biases is included in the main portion of this review.

C. Development of Input Data:

Review of older records can possibly improve the estimates of storm reccurrence rates. Cry, George W. "Tropical Cyclones of the North Atlantic Ocean", Weather Bureau Tech. Paper No. 55, U.S. Dept. of Commerce, Washington, D.C., 1965 [A-6] covers storms from 1871 to 1963, for instance.

1. Sites and Coast Segments: The coast segments chosen describe the area well and will suffice for determining the risk of hurricane strength winds at the site.

C. Development of Input Data (cont'd):

2. Occurrence Rates: The estimated tropical storm and hurricane reccurence rates are consistent with other available sources of data. The PRA evidently treats all storm as full hurricanes in the rate estimation. The occurence rate for hurricanes indicated by PRA Reference [43] is about 0.14/year crossing the shore, which is much less than 0.253/yr. assumed in the PRA. The short record for the area could be extended by search of the historical records, but the result is not likely to cause more than a 4 mph. change in the estimates for the rarer probabilities. Use of the Bayesian rate estimates is not unconservative.

3. Coast Crossing Position: The data used is reasonable and agrees with the other sources.

4. Storm Heading: The data used is reasonable and agrees with other sources.

5. Translational Speed: The data used is reasonable and agrees with other sources.

6. Central Pressure and Radius of Maximum Winds: The probability distribution for the central pressure in the PRA appears reasonably conservative. The joint distribution for the radius of maximum winds is also reasonable. The influence of any R-Δp correlation is not of great significance to the results, but is treated in an appropriate manner.

D. Hurricane Wind Risk at Indian Point:

The large number of simulations made defines the computed results quite well, with sample size influences being reduced to a negligible level. Essentially, the uncertainty lies in the wind computation procedure and the input distributions. Because of the conservative choices for occurence rates, where tropical storms of less than hurricane strength are included, as well as the reasonable or conservative choices for other parameters, the main potential for non-conservatism is in the windfield and maximum wind computations.

APPENDIX C:

These plots are associated with IV-C. They are discussed in the review of IV-C.

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A-5. Russell, Larry R. "Probability Distributions of Hurricane Wind Speeds", Report to National Bureau of Standards Center for Building Technology, Houston, Texas. April, 1979. pp. 64-68

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Figure 1	Indian Point PRA Review	7/12/82	LARRY RUSSELL & ASSOCIATES	/

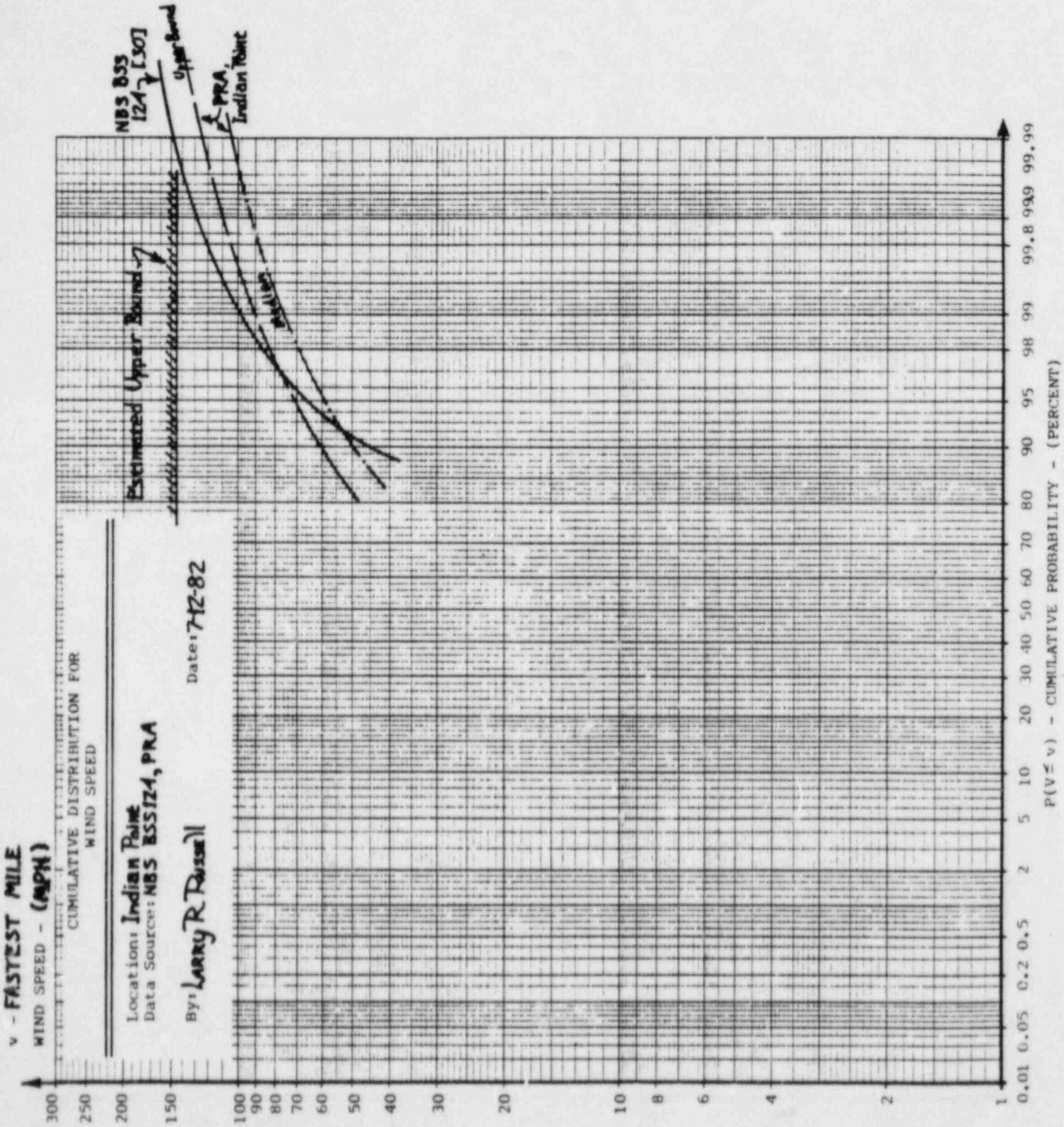

Figure 2	Indian Point PRA Review	7/12/82	LARRY RUSSELL & ASSOCIATES	/