# REVIEW OF THE INDIAN POINT PROBABILISTIC SAFETY STUDY

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## C1.0* Introduction/Overview

Early in 1982, at the request of Dr. David Okrent, several members of the staff of Argonne National Laboratory reviewed the Probabilistic Safety Study of the Zion Nuclear Power Plant of Commonwealth Edison Company, acting in the role of consultants to the ACRS. In view of the resources and staff expertise available, a few areas were selected for the focus of the review. These included:

- Basic Probabilistic Risk Methodology (Zion Sections 0, 7 and 8)
- Containment Analysis (Zion Sections 2, 3, and 4)
- External Events Phenomena (Zion Section 7)
- Existing Plant Risk Results (Zion Section 8)

The remaining sections of the report, Plant Analysis (Zion Section 1), Core Melt Accident Source Terms (Zion Section 5), Site Consequence Analysis (Zion Section 6), New Plant Feature Considerations (Zion Section 9) were not reviewed in detail. The Zion PSS review was completed in February 1982. A request to review the Indian Point PSS was received in June 1982. Since the Zion and Indian Point reactors are both Westinghouse PWR's of similar but not identical design, and the same organization led the PRA assessment, it is not surprising to see great similarities between the two PRA's. For example, the same MARCH calculations are used in both PRA's. Because of this similarity, extending to an almost identical report format, the ANL staff concentrated on two areas in its review of the IPPSS:

1. Areas where plant design differences required reevaluation of our Zion comments, particularly containment structural limit and seismic.
2. The methodology of fire analysis of Indian Point.

The reader is referred to References CR1 and CR2 for our comments on Zion. Tables C1.1 and C1.2 summarize the areas where our specific comments from Zion still are valid for the Indian Point study. In general, insufficient information was available to fully evaluate design differences between Zion and Indian Point and thus caution must be exercised. Our specific comments in the containment structural limit and the seismic evaluation of Indian Point will be presented in Sections C2.0 and C3.0. Comments on the fire analysis are presented in Section C4.0.

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*The C is added to allow distinction between comment and the Indian Point PSS section numbers.

Table C1.1
Comparison of Comments on Basic Probabilistic Safety Study Methodology

| | Zion Report (Ref. CR1) | Indian Point |
|---|---|---|
| Propagation of Uncertainties<br>a) Source term | Section C2.1* | Some changes in wording and justification but the comment still relevant |
| b) Containment matrix | Section C2.1 | Comment still holds |
| Data analysis | Section C2.2 | Comment still holds |
| Human error | Section C2.3 | Comment still holds |
| Release consequence | Section C2.4 | Comment still holds |

Table C1.2
Comparison of Comments on Containment Analysis

| Containment Node | Zion Comments (Ref. CR1) | Indian Point |
|---|---|---|
| Core Melt Incoherency (Node D) | Section C3.1* | Comment still holds |
| Steam spike/steam explosion (Node E) | Section C3.2 | Comment still holds |
| In-vessel cooling (Node H) | Section C3.3 | Based upon information contained in the report the reviewers found no significant design differences which would change our comment |
| Debris Bed Dispersion (Node I) | Section C3.4 | Based upon information contained in the report the reviewers found no significant design differences which would change our comment |
| Presence of Cavity Water (Node J) | Section C3.5 | Comment still holds |
| Basemat Penetration (Nodes K and S) | Section C3.6 | Comment still holds |
| Coolable Debris Bed (Ex-Vessel) (Node Q) | Section C3.7 | Comment still holds |
| Hydrogen Production and Burn | Section C3.8 | Comment still holds |
| Containment Pressure Structural Limit | Section C3.9 | Design differences, new comment see Section C2.0 |
| Fire Analysis | Not Reviewed | See Section C4.0 |

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*Refers to section numbers in Zion Report (Ref. CR1).

The two dominant comments relative to the Zion PSS were: 1) that the containment response treatment was not adequately documented and probably considerably underestimated the inherent uncertainties; and 2) that the source term multiplier treatment was inadequately defended. Quoting from the (Ref. CR2) ACRS Review Summary on the Zion PSS, dated April 22, 1982, which incorporates the aforementioned ANL comments,

> "Review of the phenomenology associated with the containment matrix event trees has led to some question as to whether the probabilities assigned to the branch points were in some cases optimistic with respect to both value and uncertainty range. If probabilities and attendant uncertainties assigned to these branch points have been optimistic, short cuts used in assessing the uncertainty bands may be invalid. In many cases with respect to the containment matrix, it appears that the treatment was such that no uncertainty value was assigned. Specifically, uncertainties in branch points having 1-ε or ε probabilities were ignored -- obviously this is only justified if the confidence that is implied by assigning these probabilities is justified. (ANL, C2.1)"*
>
> "By assigning probability weights to the site matrix through use of a "source term multiplier, U" the effective accident source and resulting consequence is reduced by over a order of magnitude. In light of the impact of source term reduction in the site matrix, the ZPSS authors should be requested to provide additional detail to support the probability-weighting values used. (ANL, C1.0)"*

Table C1.3 was generated to provide a handle on a results comparison between the Reactor Safety Study, Zion PSS and Indian Point PSS. The table can also be used to evaluate the relevance of the dominant Zion comments to Indian Point. The following two observations for Indian Point 2 and 3 are worthy of note:

1. The median frequencies of exceeding 100 acute fatalities for IP 2 and IP 3 are a small fraction of the core melt frequency.
2. The point values of the frequency of exceeding 100 acute fatalities are roughly two orders of magnitude larger than the "median" values.

The first point follows from the source term multiplier treatment and the analysis showing the containment to be effective in preventing a release given a core melt; the second point follows strictly from the source term multiplier. The source term multiplier treatment in the Indian Point PSS and the Zion PSS though not identical result in a similar reduction in the consequence of a release. Both PSS studies show the containment to be very effective (~ $10^{-4}$) in reducing the consequence of core melt for cases without loss of electrical power. Thus in general, our comments with respect to the methodology and execution of the ZPSS, especially as regards the need for more

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*Refers to ANL comments in Ref. CR1.

Table C1.3. Comparison Among PRA Results

| | RSS(1) | Zion(1) | IP2(2) | | IP3(2) | |
|---|---|---|---|---|---|---|
| Core Melt Frequency | | | | | | |
| Total (all causes) | | | | | | |
| Point value(3) | 4.9-5 | 5.2-5 | 47.-5 | (p. 8.3-14) | 19.-5 | (p. 8.3-15) |
| Median | | | 40.-5 | (p. 8.1-1) | 9-5 | (p. 8.1-2) |
| 90th percentile | | | 100.-5 | (p. 8.1-1) | 55.-5 | (p. 8.1-2) |
| Internal only | | | | | | |
| Point value | | 4.2-5 | 9-5 | (p. 8.5.4-7) | 13.-5 | (p. 8.5.4-25) |
| Frequency of 100 Early Fatalities | | | | | | |
| Total (all causes) | | | | | | |
| Point value | 15.-8 | 6-8 | | | | |
| Median | | 0.017-8 | 1-8 | (p. 8.5.6-2) | 0.3-8 | (p. 8.5.6-9) |
| 90th percentile | | 0.1-8 | 20.-8 | (p. 8.5.6-2) | 1.5-8 | (p. 8.5.6-9) |
| Internal only | | | | | | |
| Point value | | 0.4-8 | 4.9-8 | (p. 8.5.4-15) | 5-8 | (p. 8.5.4-33) |
| Median | | 0.0077-8 | 0.1-8 | (p. 8.5.5-12) | 0.1-8 | (p. 8.5.5-17) |
| 90th percentile | | 0.5-8 | 1.5-8 | (p. 8.5.5-12) | 1.5-8 | (p. 8.5.5-17) |

1. Taken from ANL comments on Zion PSS and Zion PSS.

2. Taken from Sections 8.5.4, 8.5.5, and 8.5.6 of IPPSS. Since figures, text, and tables sometimes yielded different values for the same parameter, the page from which the result is taken is given.

3. Point value and "mean" appear to have been used interchangeably.

through support for containment response and source term multiplier assumptions, hold for the Indian Point PSS.

A comparison between Zion PSS and Indian Point PSS obtained by inspection of Table C1.3 yields the following observations.

1. The point value of the core melt frequency from the internal causes only for Indian Point units 2 and 3 are comparable within a factor of ~ 2 to that of Zion.
2. However, the point value of the core melt frequency from all causes for IP 2 and 3 is a factor of ~ 5 to 10 greater than that for Zion.
3. The point value of the frequency of 100 early fatalities from internal causes is a factor of ~ 10 greater for IP 2 and 3 when compared to that for Zion.
4. The median value of the frequency of 100 early fatalities from all causes for IP 2 and 3 is a factor of ~ 5 to 15 greater than that for Zion.

Some of the causes of these differences between Zion and Indian Point can be highlighted by consideration of dominant sequences causing core melt. The dominant sequences for Zion, IP 2 and IP 3 are shown in Tables C1.4, C1.5 and C1.6 respectively. No single factor can be observed to cause the differences. The following observations can be obtained by comparison of these three tables.

1. External events contribute only one sequence to the top 12 sequences ranked with respect to frequency of core melt for Zion, but they contribute 5 sequences, for IP 2 and 4 sequences for IP 3. These external events have a significant effect on risk for IP 2 and IP 3 because in a significant fraction of these external event sequences the containment failure split fraction is unity.
2. The frequency of core melt as a result of the small LOCA is comparable for Zion and IP 2, with IP 3 having a slightly higher frequency (a factor of ~ 5).
3. The frequency of core melt as a result of seismic events resulting in loss of AC power is much lower for Zion ($5.6 \times 10^{-6}$), than for IP 2 ($1.4 \times 10^{-4}$) or IP 3 ($2.4 \times 10^{-5}$).

Based upon a review of the tables there is no single cause for the higher frequency of exceeding 100 fatalities for IP relative to Zion. A small part of the increase is a result of internal causes. The effect of external events, particularly events which result in loss of AC power and for which the containment is ineffective (containment split fraction of unity) have a significant effect. The external effects in question include seismic, fire and wind damage. A detailed evaluation of the plant design differences which

Table C1.4* Comparison of Core Melt and Release Frequency Contributions for Zion

| Rank with Respect to Core Melt | Sequence | Internal/ External Event | Mean Annual Frequency (Contribution to Core Melt) | Containment Split Fraction to Serious Release | Mean Annual Frequence of Serious Release | Relative Rank with Respect to Serious Release Frequency |
|---|---|---|---|---|---|---|
| 1 | Small LOCA: Failure of Recirculation Cooling | I | 1.62-5 | 1-4 | 1.62-9 | 4 |
| 2 | Seismic: Loss of All AC Power | E | 5.60-6 | 1-0 | 5.60-6 | 1 |
| 3 | Large LOCA: Failure of Recirculation Cooling | I | 4.89-6 | 1-4 | 4.89-10 | 5 |
| 4 | Medium LOCA: Failure of Recirculation Cooling | I | 4.89-5 | 1-4 | 4.89-10 | 6 |
| 5 | Loss of Main Feedwater: ATWS, Failure to Control Pressure Rise (i.e., Failure of Augmented Auxiliary Feedwater or Primary Pressure Relief) | I | 3.89-6 | 1-4 | 3.89-10 | 8 |
| 6 | Turbine Trip: ATWS, Failure to Control Pressure Rise (i.e., Failure of Augmented Auxiliary Feedwater or Primary Pressure Relief) | I | 2.76-6 | 1-4 | 2.76-10 | 7 |
| 7 | Spurious Safety Injection: Failure to Control the SI, Recirculation Cooling | I | 1.64-6 | 1-4 | 1.64-10 | 9 |
| 8 | Spurious Safety Injection: Loss of Offsite Power, Loss of ESF Buses 148 and 149 | I | 1.43-6 | 1-4 | 1.43-10 | 10 |
| 9 | Large LOCA: Failure of Low Pressure Injection | I | 1.32-6 | 1-4 | 1.32-10 | 11 |
| 10 | Medium LOCA: Failure of Low Pressure Injection | I | 4.36-7 | 1-4 | 4.36-11 | 14 |
| 11 | Loss of Main Feedwater: Loss of Offsite Power, Loss of Buses 148 and 149, Failure of Auxiliary Feedwater | I | 2.91-7 | 2-4 | 5.82-11 | 12 |
| 12 | Reactor Trip: Loss of Offsite Power-Loss of ESF Buses 148 and 149-Failure of Auxiliary Feedwater | I | 2.23-7 | 2-4 | 4.46-11 | 13 |

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*From Table II.8-22, p. II.8-87 of the Zion PSS.

Table C1.5* Comparison of Core Melt and Release Frequency Contribution of Major Scenarios, Indian Point 2

| Rank with Respect to Core Melt | Sequence | Internal/ External Event | Mean Annual Frequency (Contribution to Core Melt) | Containment Split Fraction to Serious Release | Mean Annual Frequence of Serious Release | Relative Rank with Respect to Serious Release Frequency |
|---|---|---|---|---|---|---|
| 1 | Seismic: Loss of Control or Power | E | 1.4-4 | 1.0-0 | 1.4-4 | 1 |
| 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 | E | 1.4-4 | 1.0-0 | 1.4-4 | 2 |
| 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 | E | 5.0-5 | 2.0-4 | 1.0-8 | 8 |
| 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 | I | 3.0-5 | 1.0-4 | 3.0-9 | 9 |
| 5 | Hurricane, etc., Wind: Loss of All AC Power Due to High Winds | E | 2.7-5 | 1.0-0 | 2.7-5 | 3 |
| 6 | Tornado and Missiles: Causing Loss of Offsite Power and Service Water Pumps or Control Building | E | 1.6-5 | 1.0-0 | 1.6-5 | 4 |
| 7 | Small LOCA: Failure of Recirculation Cooling | I | 1.3-5 | 1.0-4 | 1.3-9 | 10 |
| 8 | Large LOCA: Failure of Low Pressure Recirculation Cooling | I | 1.1-5 | 1.0-4 | 1.1-9 | 11 |
| 9 | Medium LOCA: Failure of Low Pressure Recirculation Cooling | I | 1.1-5 | 1.0-4 | 1.1-9 | 12 |
| 10 | Turbine Trip Due to Loss of Offsite Power: Loss of All AC Power, RCP Seal LOCA, and Failure to Recover External AC Power Until After 1 Hour | I | 6.5-6 | 1.0-4 | 6.5-10 | 13 |
| 11 | Large LOCA: Failure of Low Pressure Safety Injection | I | 5.4-6 | 1.0-4 | 5.4-10 | 14 |
| 12 | Turbine Trip Due to Loss of Offsite Power: Failure of Two Diesel-Generators, RCP Seal LOCA, and Failure to Recover External AC Power | I | 4.4-6 | 1.0-4 | 4.4-10 | 15 |

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*From Table 8.3.9, p. 8.3.21 and 8.3.22 of IP PSS.

Table C1.6* Comparison of Core Melt and Release Frequency Contributions of Major Scenarios, Indian Point 3

| Rank with Respect to Core Melt | Sequence | Internal/ External Event | Mean Annual Frequency (Contribution to Core Melt) | Containment Split Fraction to Serious Release | Mean Annual Frequence of Serious Release | Relative Rank with Respect to Serious Release Frequency |
|---|---|---|---|---|---|---|
| 1 | Small LOCA: Failure of High Pressure Recirculation Cooling | I | 8.2-5 | 1.0-4 | 8.2-9 | 8 |
| 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 | E | 6.1-5 | 1.0-0 | 6.1-5 | 1 |
| 3 | Large LOCA: Failure of Low Pressure Recirculation Cooling | I | 1.1-5 | 1.0-4 | 1.1-9 | 9 |
| 4 | Medium LOCA: Failure of Low Pressure Recirculation Cooling | I | 1.1-5 | 1.0-4 | 1.1-9 | 10 |
| 5 | Large LOCA: Failure of Safety Injection | I | 6.4-6 | 1.0-4 | 6.4-10 | 11 |
| 6 | Small LOCA: Failure of Safety Injection | I | 2.8-6 | 1.0-4 | 2.8-10 | 12 |
| 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 | I | 2.7-6 | 1.0-4 | 2.7-10 | 13 |
| 8 | Seismic: Loss of Control or AC Power | E | 2.4-6 | 1.0-0 | 2.4-6 | 2 |
| 9 | Medium LOCA: Failure of Low Pressure Safety Injection | I | 1.7-6 | 1.0-4 | 1.7-10 | 14 |
| 10 | Fire: Specific Fire in the Cable Spreading Room Causing Loss of All Control Power | E | 1.6-6 | 1.0-4 | 1.6-10 | 15 |
| 11 | Tornado and Missiles: Loss of Offsite Power and SW Pumps | E | 9.2-7 | 1.0-0 | 9.2-7 | 3 |
| 12 | Loss of Main Feedwater: ATWS and Failure of AFWS | I | 7.7-7 | 1.0-4 | 7.7-11 | 18 |

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*From Table 8.3.10, p. 8.3-25 and p. 8.3-26 of IP PSS.

cause the increase in risk for IP relative to Zion is beyond the scope of our review. However, contribution of fire-induced core melt to the overall public risk was one of the motivating factors for our review of the fire methodology.

The reviewers are concerned with regards to the confidence one can place in so-called "Level 2" curves describing probabilities of different types of damage and their attendant confidence bands. Sandia, in a separate unreviewed assessment of Zion, reported risk curves to be roughly two orders of magnitude higher than in the Zion PSS. Further, the dominant source of risk in the Sandia assessment was "internal events", not seismic as reported in the Zion PSS. Given the similarity of treatments for Zion and IP, until such differences are resolved for Zion, and until the other issues described within our comments are resolved, the reader is cautioned in the use in the actual risk probabilities displayed in the risk curves.

For IP2 and IP3 fire is an important contributor to the core melt frequency as shown on p. 8.3-28 and p. 8.3-32. The state of the art nature of fire analysis must be noted and the authors are to be congratulated for their efforts. The cable spreading room fire (a fire risk contributor) received the bulk of our review effort. The dominant comments are related to: 1) the Bayesian treatment of fire failure data; 2) the interpretation of the fire data of Fleming et al., (Ref. CR3) related to detection, application and suppression times; and 3) deficiencies in the COMPBRN model. The Bayesian treatment of fires in the cable spreading room is based upon very sparse failure data (2 fires in 301 compartment years) and a somewhat arbitrary prior. The treatment needs additional justification. The cited references reviewed interpret the available fire **duration** data of Fleming et al., in two different manners. The Indian Point authors refer to the data as fire duration the sum of (detection, application and suppression times) where Ref. CR4 referring to the same data imply it to be only the fire suppression time. Additional clarification is needed. The computer code, COMPBRN, is based upon the fire being modeled as a large, free-burning, turbulent fire. Free-burning implies that inlet or exit ventilation from the compartment has no effect on fire propagation potential and additional justification is needed.

Containment can fail to perform its function by two mechanisms: 1) excessive leakage and 2) structural failure. The present analysis by the IPPSS authors does recognize these two modes of failure; however, emphasis is placed upon structural failure. Additional emphasis needs to be placed upon leakage potential of penetrations.

The sections that were reviewed were those that match with the area of specialization of the Argonne personnel. The Argonne personnel involved in the review and their areas of specialization are as follows.

| | Area of Specialization |
|---|---|
| D. H. Cho | Molten Fuel Coolant Interactions, Debris Bed Quenching, Steam Spike |
| L. W. Deitrich | Degraded Core Phenomena |
| J. F. Marchaterre | Degraded Core Phenomena |
| T. J. Moran | Seismic Analysis |
| C. J. Mueller | Probabilistic Risk Methodology |
| D. R. Pedersen | Debris Coolability, Molten $UO_2$/Concrete Interaction |
| R. W. Seidensticker | Structural Analysis - Containment |

The comments that follow reflect the combined opinion of the members of the review group. Several people were consulted during the course of our review. These included

| Consultants Used | |
|---|---|
| W. A. Bezella | Argonne |
| D. C. Ma | Argonne |
| J. Stevenson | Stevenson and Associates, Cleveland, Ohio |

## C2.0 Review of Containment Structural Analysis for Indian Point Probabilistic Risk Assessment (PSS)

Relevant Sections of the Indian Point PSS Reviewed:

Appendix 4.4.1 "Containment Capability of Indian Point Power Plant Unit Nos. 2 and 3," by United Engineers & Constructors, Inc.

Our review of the Indian Point Structural Analysis involved reviewing the above referenced section and also a recently issued LANL report, NUREG/CR-2569, "Responses of the Zion and Indian Point Containment Buildings to Severe Accident Pressures," by T. A. Butler and L. E. Fugelso (Ref. CR5). This latter report is significant in that it predicts failure of the Zion containment building at the juncture of the cylindrical shell wall and the foundation basemat. This contradicts the conclusions reached in both the Indian Point and Zion PRAs which state that the failure pressure is limited by excessive deformation and stresses in the cylindrical shell below the springline where the seismic rebar is reduced and well above the basemat juncture. These comments are expanded in Section C2.1.

## C2.1 Indian Point Structural Limit Evaluations

Comments on the containment structural limit for Indian Point resulting from review of the Appendix 4.4.1 are as follows.

### C2.1.1 General Comments

1. The rationale and the analyses for the Indian Point containment structural analysis is clearly stated and the material is well organized.

2. Overall, there are many similarities between the approaches used in the Indian Point PSS and the Zion PSS as it pertains to containment structural integrity. While a number of the comments we had made previously on the Zion PSS apply to the Indian Point PSS there are some notable differences. One of these relates to penetrations: for Zion, analysis in this area was not presented explicitly enough to verify its appropriateness. In the Indian Point PSS, however, a much more detailed explanation is offered of the various types of penetrations which exist, e.g., electrical, large valves, cold and hot piping penetrations, and the large penetrations (equipment hatch and personnel airlocks). Since there is a general consensus as to the need for a better treatment of penetrations both as sources of excessive leakage and possible structural failure, the Indian Point PSS effort is welcomed and should be encouraged.

3. The analyst appears to have a thorough understanding about all details of the Indian Point containment structures. All the critical areas in the containment system were examined (including penetrations as mentioned above) although the extent of the analysis is very limited for penetrations and additional justification of the integrity of the penetrations is desired. The

mean yielding stress of steel (71 ksi) used in the study is consistent with the value stated in report of the LLL/DOR seismic program (Ref. CR6).

The analysis does emphasize the structural limit of the penetrations with reduced emphasis upon the leakage from the penetrations prior to structural failure. No mention was found of the difference between seals where additional pressure increases the seating force versus seals where additional pressure decreases the seating force.

### C2.1.2 Specific Comments

1. The containment for Indian Point is different from the Zion containment. The former is a normally reinforced containment, whereas the latter is a lightly reinforced, post-tensioned, prestressed concrete containment. Hence, the Indian Point containment is more susceptible to concrete cracking than the Zion containment. This has been shown in the Structure Integrity Test (SIT) on the two plants (Refs. CR7, CR8). According to the mentioned references, the Indian Point containment experienced considerable cracking at a internal pressure of 54 psi, which is much lower than for Zion containment. Hence, the concrete cracking is a primary concern for the failure mode of the Indian Point containment. Unfortunately, the role of concrete cracking incorporated in the hand calculation of Indian Point containment is not well documented, e.g., the section moment capability at the base of cylindrical wall shown in Table 3 of Appendix 4.4.1, the stresses shown in Table 2 of Appendix 4.4.1, etc. It is not clear in the report how cracking was considered in the analysis. This is important since once the crack occurs, the response of the containment may become nonlinear. The hand calculation, therefore, may not adequately reflect the actual behavior.

2. The analyst defines the containment structural limit capability as the maximum combination of temperature and pressure loadings to produce the yielding of rebars. The thermal effect is considered as an insignificant factor. See, for example page 8 of the Appendix, in which it is stated that "Accident temperature induced thermal gradients through the wall are not a factor in concrete shell design since the accident temperature effect penetrates approximately 10% of the containment thickness during the significant over-pressure phase of the accident. The cracking of the concrete shell due to containment pressurization acts to relieve secondary stresses induced by thermal gradient effects." It would be appropriate to show some calculation or to cite relevant references for supporting the above statements.

3. The calculations are based on static analysis. What would the containment capability be under the rapid loads due to very rapid hydrogen burning, i.e., impulsive-type dynamic loading?

4. What would be the results if the effects of the soil surrounding the containment were included in the analysis?

5. Section 4.3, Penetrations of the Appendix, states since no excessive cracking was found during the Structural Integrity Test (SIT) that the qualitative conclusion was reached that the penetration reinforcement was adequate and therefore these regions were not critical. This may be true, but it is becoming more and more apparent that our current knowledge of ultimate structural capability and leakage rate of penetrations is not adequate and needs upgrading. Thus greater consideration of potential leak paths of penetrations is desired.

6. As in comment 5 above similar conclusions are reached for small penetrations based on the SIT test results. It is not considered adequate to predict safe behavior at 270% of design pressure based on observations of a SIT test at 115% of design pressure. In the discussion of the evaluations of the 10 inch diameter Containment Pressure Relief butterfly values and the 36 inch diameter Containment Atmosphere Purge butterfly values (Section 4.4), it appears necessary to rely -- for some combinations of temperature and pressure -- on a "buffer pressure" between pairs of these isolation valves. Such a "backup" pressure was stated to be 52 psig. Where does this pressure come from? It seems likely that it is maintained by some dynamic system and failure of these systems could lower the capability of these valves to withstand pressure and temperature. In any event, more information should be presented.

8. It is the reviewers understanding that in the course of work being done for NRC, M. Fardis (MIT) has postulated that the extensive use of Cadwell splices in large rebars may result in some degradation of overall ductility as yielding takes place. This, of course, could lower the ultimate pressure capability of the containment. (Note: ANL does not have any formal reports of Fardis' recent work; this point was brought to our attention during informal discussions with other organizations working on containment capability.)

### C2.2 Some Comments on NUREG/CR-2569, "Responses of the Zion and Indian Point Containment Buildings to Severe Accident Pressures," by T. A. Butler and L. E. Fugelso (LANL, May 1982) (Ref. CR5).

The purpose of the comments in this section of our review is to point out some features of the referenced Los Alamos report which need to be compared to the structural integrity calculations performed for both Zion and Indian Point by the plant owners. The main point is that results presented in NUREG/CR-2569 predict failures of containment for both plants at the juncture between the cylindrical shell and the basemat. This is totally different from the owner's analysis which show failures in the cylindrical shell at a considerable distance above the basemat. It is suggested that those responsible for the calculations of Indian Point be asked to comment on this difference. We here only offer a few observations which may help to focus on resolution of these differences.

1. The Los Alamos report contains a more detailed model which probably more accurately reflects the actual structure (p. 1 of the LANL report).

2. LANL ignores thermal effects, considering them to be secondary in nature (p. 3).

3. The LANL report considers the steel liner plate to be very ductile, thus it can accommodate easily any displacements due to rebar yielding (p. 6).

4. LANL introduces nonlinear soil springs in the containment model to simulate the soil (one-way spring) and uplift effect. It is believed that this is a significant improvement over the hand calculations performed for the Indian Point PSS (p. 10).

5. LANL ignores the effect of the soil which surrounds the containment side wall (p. 10). This is probably conservative.

6. LANL assumes a normal stiffness reduction factor of 0.0001 and a shear stiffness reduction factor of 0.5 (p. 11). It is felt that variations in the shear stiffness reduction will affect not only the magnitude of stresses/displacements but the location of the critical section. Sensitivity calculations seem prudent and are recommended. (Note: On p. 13, LANL states that while they do not include the effects of reinforcement ties or stirrups such effect is indirectly included by assuming a significant amount of shear stiffness after cracks develop.)

7. The 1% reduction factor on normal stiffness of cracked concrete may be unduly conservative (p. 19).

8. LANL maintains (correctly -- we believe) that uplift of the basemat is very important (p. 21).

9. As mentioned above the assumption of a 50% shear reduction factor should be tested by a sensitivity analysis (p. 22).

10. At an internal pressure of 118 psig for the Indian Point containment, the LANL analysis predicts a concrete crack approximately 3/4 of the concrete wall thickness at the intersection of the wall and basemat (p. 25).

11. Meridional rebars appear to yield initially at an internal pressure of 105 psig (p. 29), which of course is significantly less than the 126 psig value given for the estimated structural limit. Note the IPPSS analysis defines the structural limit as the pressure at which yielding of the rebars occurs.

## C2.3 Summary of Containment Structural Limit

The failure pressure of the Indian Point containment structure is predicted to be 126 psig (2.7 times the design pressure). The uncertainty associated with this limit is stated to be very low, less than ± 2 psia within a 1 sigma uncertainty band. Significant items that may affect either or both the predicted failure limit and the uncertainty are related to:

1. Failure limit and leakage from penetrations.
2. The uncertainty of concrete cracking upon the ultimate limit which was not included in hand calculation of Indian Point.
3. The penetrations of note for special concern are the butterfly values (Section 4.3 of the Appendix) that require a buffer pressure to prevent failure. The question relates the consideration given to the possible failure of the system that maintains this pressure.

## C3.0 Review of Seismic Initiating Events for the Indian Point Probabilistic Safety Study

Relevant Sections:

a) Section 7.2 Seismic
b) Appendix 7.9.1 Seismic Ground Motion Hazard at Indian Point Nuclear Power Plant Site.
c) Appendix 7.9.2 A Seismic Exposure Study for the Indian Point Nuclear Generation Station.
d) Appendix 7.9.3 Structural Mechanics Associates Inc., Fragility Study.

In reviewing the seismic related risk the reviewers from Indian Point were guided by the following:

1) Probabilistic risk analysis including seismic initiating events is a new field; the only comparable study is that of the Zion plant and it was conducted by essentially the same people. Thus independence of the review process and a broad overview are called for rather than detailed verification of calculations.

2) The relative importance of seismic initiating events to other initiating events is as important as the absolute values determined by the study.

With these factors in mind we asked the following questions: Is the procedure used valid and complete? Are the assumptions used reasonable, clearly documented, and consistent both within the report and with other available data? We reviewed both the seismicity assumptions and the fragility calculations. We did not attempt to review the seismic related plant logic (fault trees). The following observations arose.

1. Two consultants on seismicity were used and their recommendations differ markedly concerning the probability of sustained peak ground acceleration above about 0.3 g. The study used a compromise between the two recommended seismicity families which had a 0.483 probability of no effective peak ground acceleration above 0.3 g and 0.885 probability of no peaks above 0.5 g. This strongly effects the resultant probability of core melt due to seismic initiating events and thus influences the relative importance of seismic events to overall plant risk. In the Zion PSS, the median seismic core melt frequency was $2.0 \times 10^{-6}$ whereas that value was $9.3 \times 10^{-5}$ for Indian Point Unit 2 and $2.4 \times 10^{-8}$ for Indian Point Unit 3. While the key fragilities of the plants differ somewhat, in particular Indian Point Unit 2 is quite susceptible to peak acceleration near 0.3 g, it is our estimate, without performing the detailed fault free analysis, that the two order of magnitude difference between the core melt frequencies for Zion and Indian Point Unit 3 is primarily due to the seismicity families chosen. No discussion of this was found in the report and it is our opinion that this is a weakness in the study.

2. Sections 7.2.4.1 and 7.2.5.1 consider the collapse of a masonary wall in the control rooms and dismiss it from further consideration because the control cabinets would only be dented. The effect of this high stress situation on operator performance needs additional evaluation. Section 0.19.3 indicates that high stress human failure rates are treated on a case by case basis.

3. The use of a log normal distribution for describing uncertainty and variability in fragilities is probably justified although the assertion that such distributions are accurate to the 0.01 level is optimistic. There is evidence that log normal distributions show considerable variation with failure data at the 0.05 to 0.10 level. This becomes significant because, as is pointed out in the report, the seismic risk is associated with an interaction of the tails of the seismicity distribution and the fragility distributions. Further study of this issue is warranted.

4. The distinction between uncertainty and random variation in fragilities is clear within the text of the report but in section 7.9 where the calculations are illustrated what is calculated is a total logarithmic standard deviation, $\beta_c$, for the fragility. This is then, often times arbitrarily, divided into contributions due to random variation, $\beta_r$, and uncertainty, $\beta_u$. As a result the distinction between these parameters is overemphasized in the body of the study.

5. The fragility calculations are documented only by generic method and example. For most of the fragilities only the results are given in a table and the specific assumptions are not available. Apparently there is less seismic margin in the Indian Point structures than in Zion. The component fragilities are roughly comparable between the studies. Their is no way to conduct a verification of these fragility calculations without considerable effort and access to the detailed design reports but because all the fragility calculations for Indian Point, Zion, and the SSMRP are done by one organization, it is important that such a check be performed by a third party.

6. It is not clear from this report whether the personnel involved in calculating the fragilities of the Indian Point structures and components actually visited the site and inspected the buildings and components. Such site visits are extremely important in identifying factors such as conditions of buildings and components and relationships between the failure of non critical components and safety related components. If such visits were not conducted the uncertainties in the fragilities should be modified accordingly.

## C3.1 Conclusion Relative to Seismic Risk

While seismic risk studies of this kind are a new field and should be judged accordingly, a deficiency in this report is the failure to address the implications of the differences of opinion on the probability of high acceleration seismic events. It appears to this reviewer that both the magnitude of the risk to the public and the relative importance of seismic events to that risk are strongly dependent on the seismicity source assumptions. A great deal was learned about the effects of earthquakes on the Indian Point plant which could lead to modifications to improve its seismic resistance.

## C4.0 Contribution of Fires to Risk

Relevant Sections:

a) Section 7.3 Fire
b) Section 8.3 Identification of Major Contributors to Risk

The Indian Point PSS identified that fires are a significant contributor to public risk as is evident by review of Tables 8.3-9 and 8.3-10. Our review of the Indian Point Fire Analyses plus supporting references will concentrate on three areas:

1. Basic procedure for fire risk evaluation.

2. Evaluation of cable spreading room fires for Indian Point 2.

3. Evaluation of the computer code COMPBRN.

4. Additional comments.

The analysis of fires present in the IPPSS is very extensive and difficult to review. The cable spreading room fire for Indian Point 2, being the first location presented in the report and important to risk, received the most of our review effort.

## C4.1 Basic Procedure for Fire Risk Evaluation

Relevant Sections:

a) Section 7.3.1.1 Summary

The basic procedure for evaluation of fire ana[illegible]dicated by the authors involves a six step process. The authors sta[illegible]p 1 is the identification of locations that require more detailed analysis. The authors criterion for preliminary selection of a critical areas was that the qualification of "an area as critical is that a fire occurring in that location must be capable of causing an initiating event (a LOCA or a transient)". It is our opinion that this preliminary criterion for critical areas should be expanded to include these areas in which occurance of a fire may lead to the requirement that the reactor be shutdown. Then given that this postulated fire would cause either 1) an initiating event (LOCA or transient), or 2) a requirement to shutdown the reactor, proceed to determine whether the same fire could induce failure that would prevent:

1. Reaching and maintaining a condition of negative reactivity.

2. Removing core decay heat.

3. Monitoring and controlling the primary system coolant inventory.

Examples of the the second type of fire location are: 1) control room fire, or 2) a cable fire of the Brown's Ferry type.

The report identified the following critical areas:

1. Cable spreading room.
2. Switchgear room.
3. Two portions of the electrical tunnel.
4. The Motor Control Center area.
5. The electrical penetration area.
6. Diesel generator building.

Why was the control room not classified as a critical area? An explanation of why the control room was not considered would be desirable.

## C4.2 Evaluation of Cable Spreading Room Fires for Indian Point 2

Relevant Sections:

a) Section 7.3.1.2 Cable Spreading Room (Fire Zone 11, Control Building)

Our review will concentrate on two areas: 1) the characteristics of the room, and 2) the analysis of the frequency of core melt as a result of fires.

### C4.2.1 Characteristics of the Room

The characteristics of the room are discussed in Section 7.3.1.2.1, p. 7.3-4. The characteristics of the room noted in our review having impact on the fire risk are as follows:

1. It is stated that the combustibles stored in the room consist of cable insulation, motor lubricant, paint and transient fuels. The total heat content of the room is given as 527,000 BTU* with approximately 80% of this total in transient fuels. Why is not the cable spreading room devoted to a single purpose, i.e., spreading cables, and why is it necessary to use this room to store transient fuels? Would not the risk from fires be reduced if the amount of combustibles were reduced. The report and method of analysis considers the type of combustible in the analysis of propagation but only indirectly considers the amount of combustibles.

2. The cable spreading room (see p. 7.3-6, Sect. 7.3.1.2.1, No. 3) is connected to the electrical tunnel. A manual Halon system is sized to cover the volume of the spreading room and the connected portion of the electrical tunnel. The report listed as a positive issue that the electrical tunnel ventilation fans are arranged such that smoke from a fire would be withdrawn from the cable spreading room in the opposite direction from which the fire brigade would be entering to fight the fire. If

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*Equivalent to approximately the energy delivered from ~ 45 lb of Illinois coal.

the ventilation fans exhaust to outside the volume of the cable spreading room and the electric tunnel, what happens to the Halon concentration? If all the openings to the cable spreading room have shutters, louvers, or dampers that are automatically closed upon Halon system activation, when and how are the exhaust fans activated and how effective are they in removing smoke from the cable spreading room? Because of the importance of the ventilation system to fire suppression, consideration should be given to incorporating the failure of this system into the analysis. The operational modes of this system may not necessarily be included in the event trees but perhaps the human error or operator effectiveness parameters could be a function of the ventilation system performance. At the very least, some further discussion of ventilation operation is required. In addition, consideration of the thermal load placed upon the ventilation system and its operational limits is needed.

3. The reactor control room is located over the cable spreading room (see p. 7.3-5). The cable penetrations from the cable spreading room enter the region of the safeguards supervisory panels in the control room. What is the method of sealing the penetrations? Is smoke build-up in the control room possible to the point of requiring evacuation of the control room?

The implication of these room characteristics upon the parameters chosen in the analysis are unclear and further clarification is warranted.

### C4.2.2 Analysis of the Frequency of Core Melt as a Result of Fire

Relevant Sections:

a) Section 7.3.1.2 Cable Spreading Room (Fire Zone 11, Control Building)

The frequency of core melt as a result of fires in the cable spreading room is given by (see p. 7.3-13)

$$\phi_{CM,CSR} = \lambda_{CSR} f_{es} Q(\tau_G) Q_S$$

where

$\lambda_{CSR}$ = frequency of fires in the cable spreading room.

$f_{es}$ = fraction of cable spreading fires that are large and near the center of the northern wall.

$Q(\tau_G)$ = conditional frequency of fire growth.

$Q_S$ = frequency of failure to activate the auxilary feedwater system.

The discussion will be further divided on the basis of each of these terms.

## Frequency of Fires

The first term, frequency of fires in the cable spreading room, $\lambda_{CSR}$, was taken from Ref. 7.3.3* (also in Ref. CR4 and CR9). The frequency of fire is determined using Bayes technique and is dominated by the evidence of cable spreading room fires. The available evidence (as of May 1, 1978) of 2 cable fire spreading room fires in 301.3 compartment years is both sparse and somewhat questionable. It includes the Brown's Ferry cable tray fire and based upon review of Ref. CR3, Table 3-1, Event #18 a fire at Millstone 2 in 1977 (interpretation of table is difficult and verification is necessary). The fire at Millstone appears to have occurred in the cable tray caused by ignition of oily rags in the cable tray by welding sparks. Mention is made several times in the report of the overly conservative use of the Browns Ferry fire in the data base (e.g. pages 7.3-3 and 7.3-6) based on improvement in procedures and/or design that resulted from that fire. This is somewhat misleading and should be clarified based on the following comment: It is our understanding that the Browns Ferry fire should be classed as a direct fire within a cable tray whereas it is included in the Indian Point data base for exposure fires in the cable spreading room. Fires that start within the cable trays were not considered by the Indian Point PSS authors based on height considerations and the lack of power cables in this area. Therefore it appears that the Browns Ferry fire was not actually modeled in this instance but used merely as a data point for estimating the frequency of occurrence of a less severe exposure fire. A similar comment could be made with respect to the Millstone cable tray fire based upon our interpretation of the data of Ref. CR3. Thus it appears that the fires that have occurred in the cable spreading room have been direct fires (in the trays) but the modeling of growth using COMPBRN is for exposure fires (fires on the floor).

## Fraction of the Large Cable Spreading Fires

The conditional frequency of large fires in the cable spreading room, $f_{es}$ was assumed to be lognormal with the characteristic values

$$f_{es,05} = 2.5 \times 10^{-4}$$

$$f_{es,50} = 5.0 \times 10^{-3}$$

$$f_{es,95} = 0.1$$

$$\alpha f_{es} = 2.6 \times 10^{-2}$$

It is stated: "The distribution for $f_{es}$ is assessed to be lognormal with the following characteristic values:." What is the basis for selecting the numerical values for $f_{es}$? This appears to be but one of a number of numerical values chosen by the IPPSS authors with very minimal justification.

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*IPPSS Report Reference Number.

In our assessment of these values and the need for additional justification we point out in the following:

1) The pilot fire chosen for propagation analyses using COMPBRN was a 1 ft diameter oil fire. This represents a large combustible source. Would it not seem reasonable to use the code to also predict the effect amount of combustibles and the available area for the fire in the development of the distribution function for large fires. Several calculations of this nature are reported in the analysis of propagation times (Section 7.3.1.2.3.2) but the use of these calculations to justify the frequency of larges fires is not presented.

2) Fleming, et al., Ref. CR3, reviewed the available data on fires and presented the complimentary cumulative distribution as a function of fire diameter. This graph is reproduced in Fig. C4.1. The horizontal cross section of the cable spreading room in Indian Point 2 is 8 ft high, 10 ft wide. One classification of a large fire is fires whose fire diameter is greater than a characteristic length of the room, say 10 ft. Approximately 25% of the fires experienced the damage diameter exceeding 10 ft. How does this compare with the distribution chosen.

Conditional Frequency of Fire Growth $Q(\tau_G)$

The conditional frequency of fire growth requires a calculation of the time for fire growth propagation and cable failure balanced against the time for detection of the fire, application and suppression of the fire. Application is the time between detection and inception of suppression of activities. The conditional frequency of fire growth, Q(t), is defined to be the conditional frequency that the fire is not extinguished by time t, given a fire or

$$Q(t) = \exp(t/\tau_D)$$

where

$t$ = time for fire growth and cable failure

$\tau_D$ = fire duration, equal to a sum of $\tau_{DE} + \tau_A + \tau_S$ (time for detection $\tau_{DE}$, time for application of fire fighting efforts, $\tau_A$, and time for suppression, $\tau_S$).

The time for fire growth is calculated by use of COMPBRN. A review of COMPBRN is presented in Section C4.3; however, a few comments are in order at this time. In the scenario chosen, the ground base exposure fire first ignites the lowest cable tray after $\tau_1$ minutes, spreads to the critical trays after $\tau_2$ minutes. The values $\tau_1$ and $\tau_2$ were calculated using COMPBRN. To quantify the uncertainty in COMPBRN, the authors modified these with two lognormally distributed parameters, $\psi$ for $\tau_1$ and $\phi$ for $\tau_2$. The logic used in the assignment of the numerical values to $\psi$ and $\phi$ was not clear to the readers and the additional clarification by the authors would be desirable.

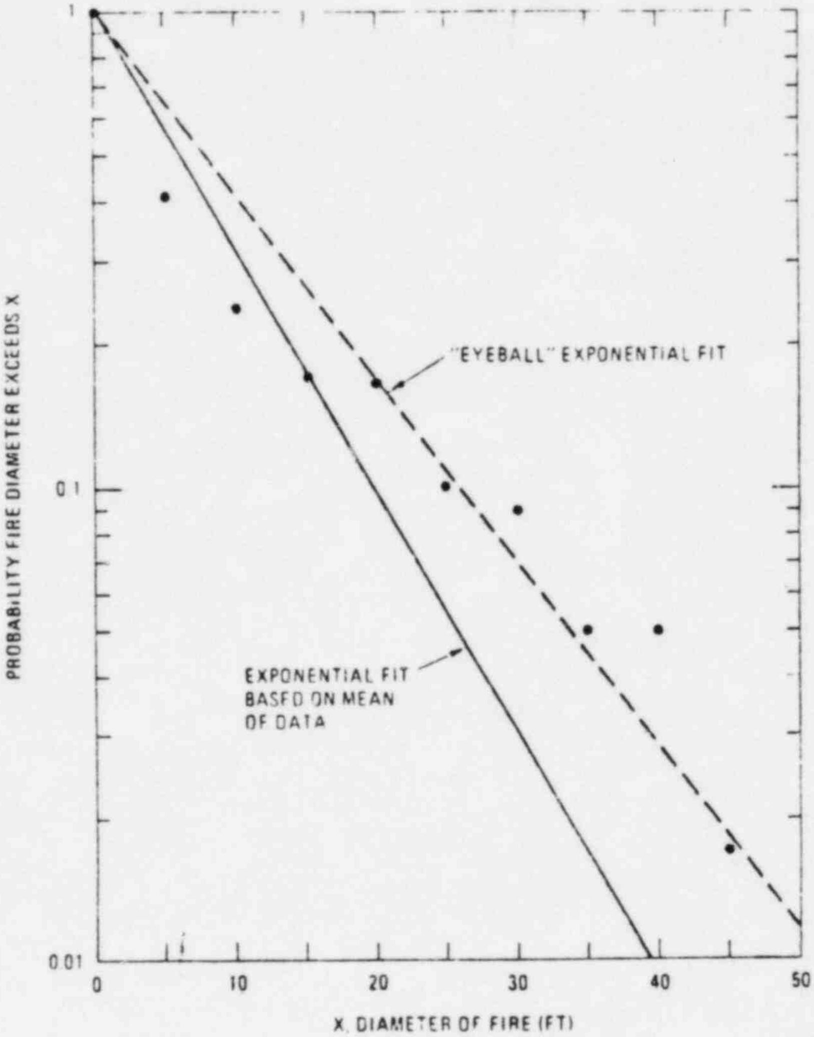

Figure C4-1 Exponential Fits of Fire Diameter Complementary Cumulative Distribution Function.

From Fleming Ref. CR3.

To evaluate the conditional frequency of fire growth you need to know the time for detection ($\tau_{DE}$), application ($\tau_A$), and suppression ($\tau_S$). We will begin our review with a discussion of the detection time, $\tau_{DE}$. The authors use the report of K. N. Fleming, et al., (Ref. CR3) and an additional unreferenced report published at approximately the same time as the Indian Point Report (Ref. CR4). Fleming, et al., surveyed U. S. nuclear plant information for abnormal occurrences to obtain fire data for U. S. plants up to May 1978. The authors of the Indian Point report state on page 7.3-9 that

> ...The representative detection time (including remote and local combined) for the fires considered in the preceding section is on the order of a few minutes and the uncertainty range may vary from a few seconds to approximately 15 minutes. The latter bound is the sum of fire propagation time 8 minutes and the time before the operators realize that the cause of the control room abnormalities is a fire in the cable spreading room....

These times are very short and additional justification is desirable. In the case of fires caused during the process of plant maintenance or modification such as Brown's Ferry or at Millstone this low time appears more than reasonable for detection because of the proximity of plant personnel. However, the abnormal occurrence reports surveyed by Fleming, et al., attempted to determine the time for detection of the fire. Of the approximately 50 fires during reactor operation surveyed in the report, only 5 included an estimation of the time between ignition and detection, (detection time $\tau_{DE}$). These involved estimated detection times from 20 minutes to 3 hours (see Ref. CR3 or Ref. CR4). The effect of this data upon the detection time chosen by the authors of the Indian Point PSS is uncertain and additional information should be requested. No discussion in the Indian Point PSS of the application time ($\tau_A$), the time between detection and initiation of efforts to suppress the fire, was found by the reviewers in the report.

The final time of concern is the fire suppression time. Kazarian and Apostolakis (Ref. CR4) in their evaluation of the data base gathered by Fleming, et al., make the following comment on page 163:

> ...The data* includes a column entitled "Time to Bring Fire Under Control". Based upon the numbers of that column we have derived the two frequency distributions of Table D.1**; one uses all the cases that occurred when under commercial operation, the other used those cases that involved electrical insulation (also under commercial operation). We believe the numbers in that column represent the time between fire detection and fire growth inhibition. It is after the detection time because the adjacent column is entitled "Interval Prior to Detection" and in some cases, it contains numbers

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*Refers to Table 3-1 of Ref. CR3.

**Attached as Table C4.1.

## Table C4.1 From Ref. CR4

### TABLE D.1 FREQUENCY DISTRIBUTION OF "TIME TO BRING FIRE UNDER CONTROL" FROM REFERENCE D.1 FOR THE COMMERCIAL OPERATION PHASE

| Time | All Cases | | Cases Where Electrical Insulation was Involved | |
|---|---|---|---|---|
| | Frequency (%) | Cumulative Frequency (%) | Frequency (%) | Cumulative Frequency (%) |
| ~ 0 | 10 | 10 | 12 | 12 |
| 3 min | 8 | 18 | 24 | 36 |
| 6 min | 25 | 43 | 12 | 48 |
| 12 min | 17 | 60 | 6 | 54 |
| 18 min | 9 | 69 | | |
| 30 min | 11 | 80 | 23 | 77 |
| 42 min | 4 | 84 | 6 | 83 |
| 1 hour | 4 | 88 | 6 | 89 |
| 1-3 hours | 4 | 92 | 6 | 95 |
| 1-5 hours | 2 | 94 | | |
| 2 hours | 2 | 96 | | |
| 7 hours | 2 | 98 | 5 | 100 |
| 24 hours | 2 | 100 | | |

> larger than the column. Thus the distributions of Table D.1* depict a period that is certainly smaller than the fire duration**. Also it should be noted that the time periods given in Ref. D1*** are, in most cases, estimates by experts and not the results of actual time measurements....

The authors of the Indian Point PRA referring to the same data of Ref. 7.3.6 (Ref. CR3) make the following statement:

> ...Reference 7.3.6 estimates the duration of actual fires. This**** duration is defined as the time from ignition to the time when maximum damage has occurred. The estimates of Reference 7.3-6, which are the results of expert evaluation, demonstrated that, of the 18 fires studied, approximately 80% were extinguished manually. The experts estimated that seven (40%) of them were extinguished within 5 minutes. Four of the 18 fires were estimated to have been extinguished between 5 and 25 minutes after initiation, and another group of four fires had extinguishment times of between 30 and 35 minutes. There were two fires that were extinguished after 60 minutes but before 85 minutes. Finally, the Browns Ferry fire was extinguished in approximately 7 hours....

In the interpretation of what appears to be the same data, one reference appears to refer to the data as the fire duration (the sum of $\tau_{DE} + \tau_A + \tau_S$) where the other reference appears to refer to the data as just the fire suppression time $\tau_S$ (detection and application time not included). This same apparent confusion to the reviewers seems to exist within Ref. CR4 (see pages 100-103 and page 163). Further elucidation by the authors of the Indian Point PRA relative to their interpretation of the data of Ref. 7.3-6 and the development of the distribution of fire duration given in Table 7.3.1-1 should be requested.

### C4.3 Evaluation of the Computer Code COMPBRN (The Fire Propagation Model)

Relevant Section:

a) Section 7.3.4 Appendix I - Fire Propagation Model

A model for estimation of the fire propagation time is presented. The authors are to be complimented for this development. The evaluation is, however, difficult because of the limited information presented. The section refers the readers to Ref. 7.3.4-1 (N. Siu, UCLA-ENG 8090, University of California, Los Angeles, 1980) for additional information. Attempts to obtain

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*Attached as Table C.1

**Sum of $\tau_{DE} + \tau_A + \tau_S$.

***Ref. D1 is Ref. 7.3.6 of the IPPSS.

****Underline added.

this reference from the library were unsuccessful. Contact with the author indicates the report is being rewritten and will be released in the near future. The following comments can be made concerning the model.

The model is characteristic of a large, fully-developed, free-burning fire. A fire of combustibles will usually consist of three phases, a growth period marking the time for flash over, fully-developed period in which the combustibles undergo flaming combustion and the maximum temperatures are achieved and finally a decay period in which the temperature fall due to source depletion with no suppression efforts. The growth period is short relative to the fully-developed period, typically less than 10 minutes (see Ref. CR10, p. 40-41). Modeling in the growth phase is extremely complicated and not fully understood at this stage. The assumption of fully-developed fires is conservative in that shorter times for propagation are predicted. The remaining assumption is that fire is free-burning. A free-burning fire is a fire in which the air-supply (ventilation) to the fire doesn't control the flame temperature and/or burning rate. In addition, the model seems to imply no restrictive ventilation of the hot combustion gases from the compartment.

The assumption of free-burning may not be conservative. The two methods of energy transfer from the flame to an object in the room are by radiation and by convection. For objects with line of sight contact radiation may be the dominant mode of energy transfer. The authors have stated in the write-up that 20-40% of the energy transfer is by radiation. This assumption is correct for luminous flames but incorrect for non-luminous flames for which the radiation transfer from the flame may be limited to less than 10% (see Refs. CR10, CR11). Examples of luminous flames are flames from methyl alcohol, benzene, kerosene, liquid n-butane, wood, and liquid hydrogen and correspondingly, examples of material with non-luminous flames are methanol and acetone. Thus the type of combustible material can have a strong impact on the fraction of the energy transferred by radiation. The combustibles in the room, both transient and permanently present, should be classified prior to the analysis. The fraction of the energy received by the object is determined by the radiation shape factor from the flame to the object. No reference is given relative to the flame height to diameter correlation used. Additional information should be requested.

Convection heat transfer may well represent the dominant mode of energy transfer from the flames to objects not in direct line of contact to the fire source. The reviewers were unable to determine the exact treatment of convection heat transfer from the fire based upon the available information. The gas temperature as a result of a compartment fire may be fairly uniform in an enclosure except near the location where cool air enters. The authors seem to imply that the temperatures are a function of the elevation above the source which would be characteristic of a buoyant plume above a free-burning fire. The reviewers were unable to determine the exact treatment of the effects of ventilation, both inlet and outlet from the compartments, upon the maximum gas temperatures in the enclosures.

The gas temperatures achieved in a compartment fire are a strong function of the ventilation areas (both inlet and outlet). The average gas temperature in an enclosure can be reduced considerably if the inlet ventilation is reduced to the point to limit oxygen availability. For inlet ventilation areas above a certain size the flame is no longer oxygen limited and the maximum gas temperature in the compartment is determined by the exit ventilation area. For large inlet and outlet ventilation area relative to the fire size the fire may be modeled as a free-burning fire with a buoyant plume. The authors are referred to the Reference CR12, p. 68 for one treatment of the temperature/time history of fire in an enclosure (compartment) including the effects of inlet and exit ventilation. The analysis of Ref. CR12 does not consider heat transfer to objects in the room but modification should be possible. It does present a method of consideration of the limited amounts of combustibles in the room in limiting the maximum gas temperature. An item to be noted is that if the gas temperature in the room is uniform, which it would be for fires in which the exit ventilation controlled, then the dominant mode of energy transfer is by convection and the definition of critical areas for fire occurrence in a compartment are less meaningful.

The requirements for cable failure in the analysis are that the temperature of the cable reach the ignition temperature of the cables or ~ 550°C (see page 7.3-13). The calculation of the temperature is based on the analytical solution of the surface temperature of a semi-infinite media with a uniform surface heat flux. This technique is excellent for cases where the dominant mode of energy transfer is by radiation and the heat fluxes are large. For convection dominated fires or for long ignition times relative to thermal time constant of the cable the finite size of the cables needs to be considered in the analysis.

### C4.3.1 Summary of COMPBRN

The authors of the Indian Point PSS in the use of COMPBRN to model fires have assumed a large, fully-developed, free-burning fire. The reviewers believe this position is justified if the ventilation areas (inlet and exit) are large relative to the fire burning rate preventing either oxygen starvation or restricted venting of hot gases from the fire. The authors should be requested to provide further justification of this position. If ventilation is important a method of calculation is recommended for consideration (see Ref. CR12, p. 68). Consideration of the differences between luminous and non-luminous flames in terms of the effect on radiant energy transfer are recommended. It is recommended that the model be expanded to consider the finite quantity of combustibles in the room as the propagation potential of a fire is a function of the quantity of combustibles in the room. The PRA analysis should recognize the quantity of combustibles stored in the room (a utility with excellent fire prevention practices should receive the credit deserved). The finite size of cables should be considered in the determination of the time to reach the ignition temperature.

### C4.4 Additional Concerns Relative to Fire Risk

#### Common Mode Equipment Failures

A major concern with fires and an important consideration is the potential for a treatment of common mode failures. The Indian Point fire analysis appears to include a comprehensive assessment of the multiple disablement of safety equipment. However, it might make this treatment of common mode failures more visible by at least some mention of common mode failures in the introductory summary section. A discussion of how the probabilistic analysis is affected when common mode failures are considered would be helpful in understanding the subsequent sections.

### Human Errors

Several accident sequences require the active intervention of the operators (e.g. p. 7.3-14 and 7.3-15). Since human error is a major contribution to the frequency of failures in several events (e.g. p. 7.3-15) some additional support for the reasonableness of the human error parameters selected and sensitivity of the predictions to these values would be helpful.

### Frequency of Fires in Cable Spreading Room Versus Electrical Tunnel

On page 7.3-18 the statement that the frequency of the electrical tunnel fires being lower than that for cable spreading room fire based on a previous observation is not clear and needs elaboration.

### Fire Risk-Indian Point 2 Versus Indian Point 3

Some discussion of the difference in the Indian Point 2 and Indian Point 3 fire results is required. In particular, the reason for the higher Indian Point 3 frequency of containment event tree entry state TE. This result does not seem to be consistent with the overall improvement noted for Indian Point 3 fire results over the Indian Point 2 results.

### References

CR1. Letter, R. Avery, ANL, to D. Okrent, ACRS, "Review of Zion PSS", February 17, 1982.

CR2. Letter, ACRS Review Summary of the Zion PSS, April 22, 1982.

CR3. K. N. Fleming, W. J. Houghton and F. P. Scaletta, "The Methodology for Risk Assessment of Major Fires and its Application to an HTGR Plant," General Atomic Company Report GA-A15402, UC-77, July 1979.

CR4. M. Kazarian and G. Apostolakis, "Fire Risk Analysis of Nuclear Power Plants," University of California, Los Angeles, Report # NUREG/CR-2258, also UCLA-ENG-8102, 1981.

CR5. T. A. Butler and L. E. Fugelso, "Responses of the Zion and Indian Point Containment Buildings to Severe Accident Pressures," LANL report, NUREG/CR-2569, May 1982.

CR6. P. D. Smith, O. R. Masleuikov, and S. E. Bumpus, "LLL/DOR Seismic Conservatism Program Investigations of the Conservatism in the Seismic Design of Nuclear Power Plants," UCRL-52716.

CR7. "Structural Response of the Containment Structure of the Zion Nuclear Plant Unit No. 1 During Structural Integrity Tests -- Commonwealth Edison Company," for Sargent and Lundy (submitted by Wiss, Janney, Elstner, and Associates, Inc.) (March 1973).

CR8. "Report on Consolidated Edison's Indian Point Unit No. 3 Containment Vessel Structural Integrity Test," for WEDCO Corporation (prepared by United Engineers and Constructors Inc.) (February 1975).

CR9. G. Apostolakis, M. Kazarian and D. C. Bley, "Methodology for Assessing the Risk from Cable Fires," Nuclear Safety, Vol. 23, No. 4, p. 391-407, July-August 1982.

CR10. M. P. Murgai, "Similarity Analysis in Fire Research - With Special Reference to Fire Convection Inside Enclosures," Oxford and IBH Publishing Co., New Delhi, 1976.

CR11. Kent's Mechanical Engineer's Handbook, Power Volume, J. Kenneth Salisbury, Editor, p. 2-65, 12th Edition.

CR12. M. P. Murgai, "Natural Convection from Combustion Sources," Oxford and IBH Publishing Co., New Delhi, 1976.