March 12, 1997

Casimir A.Kukielka
3926 Walbert Ave.
Allentown, PA 18104

Eric R. Jebsen
3213 East Blvd.
Bethlehem, PA 18017

Mary Drouin
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
MS T10E50
Washington, DC 20555

Dear Ms. Drouin:

Please find enclosed comments on the draft NUREG 1560. These comments reflect the views of the authors of this letter and in no way reflect the comments of the Pennsylvania Power and Light company. We appreciate the opportunity to comment on this report.

Sincerely yours,

Casimir A. Kukielka

Eric R. Jebsen

Casimir A. Kukielka P.E.   Eric R. Jebsen P.E.

01/1P

Comments on NUREG-1560, Vol. 1, Draft

Comments on Executive Summary:

1. The report should emphasize that it is a survey of the results of the IPE submittals. The results of the IPEs reflect the disparate views and multiple approaches of PSA practitioners. This variability in PSA practice reflects the state of the art of PSA, a state that continues to evolve given the relative youth of PSA. The lack of uniformity of results also reflects the differences in plant design and operation, and echoes the conclusions of NUREG-1150 whose underlying studies were originally an attempt at "updating" WASH-1400 to determine operational risk across the nuclear industry, but which concluded that broad application of results was not possible after study of five individual nuclear plants. Each IPE is a snapshot of both the state of the art of PSA and a particular plant's equipment, procedures, and training at the time the IPE was performed. Results of subsequent examinations can be expected to differ from those presented in the IPEs as PSA evolves and the improvements identified in the IPEs are implemented at each site. Given the improvements identified in the supplements to the generic letter and those identified at each site, one would expect a substantial improvement in the calculated core damage and containment failure frequencies in future risk calculations.

2. The report should also state that there were four purposes of the IPE process, not just identification of plant specific vulnerabilities. That is, the appreciation of severe accident processes and understanding of the most likely severe accidents at a particular site are equally valid in developing and enhancing safety culture. What can be agreed on is that the "bottom line" numbers coming out of a PSA are perhaps its least useful result. It is the systematic examination and understanding process of PSA which is its strength. Given that most plants identified improvements in equipment, procedures, and training, the IPE has improved operational plant safety.

3. The executive summary states (p. xviii) that ". . . before an IPE/PRA can be used beyond its original purpose . . . the quality of the IPE/PRA will need to meet standards . . ." It should be pointed out that currently a plethora of PSA approaches exist, and that "quality" and "standards" for PSA are only in the most rudimentary form. While it is agreed that "quality" is desired, the definition of "quality" is far from complete. Waiting until perfect "quality" of PSA is achieved before utilizing the results is impractical. The existing IPEs form the basis of compliance with such regulatory requirements as the Maintenance Rule and G. L. 89-10 valve ranking, despite the lack of "quality" guidelines for these applications. It is expected that "quality" and "standardization" will evolve, not through *a priori* definition, but through frequent, repeated <u>application</u> and peer review of PSAs.

4. Page xx of the summary presents overall conclusions and observations. Given that the primary objective of the IPE program is the identification of plant specific vulnerabilities (pg. xi of the summary), the extent to which the program accomplished this objective should be stated in the "Overall Conclusions and Observations" section.

5. The report represents a comparison of all the IPEs performed. The Susquehanna IPE was performed as an "other systematic method" and cannot be directly compared to a conventional PRA. The Susquehanna approach incorporates deterministic criteria for equipment, procedures and operator interfaces into the examination process. This was specifically aimed at addressing the extreme sensitivity of the calculated plant damage frequencies to assumptions concerning operator error, common cause failure, and core damage progression.

6. As described in comment 5, the PRA model as currently exercised lacks structural stability. Here structural stability is defined as a model whose output is not significantly changed by perturbations to the input or to the model itself. The results of current PRA's in terms of both calculated plant damage frequencies and the insights gained from the PRA are very sensitive to common cause failure and human error inputs. This weakness in the PRA model cannot be addressed via quality assurance or imposition of consistency in particular inputs since the problem is not one of quality but of epistemology. This lack of structural stability in the current PRA model, due to these inputs, limits the utility of the PRA especially as a regulatory tool and may in fact have a deleterious impact on overall safety.

This lack of structural stability was addressed in the Susquehanna IPE by requiring that all accident sequences, regardless of their frequency, satisfy Defense in Depth in terms of equipment, procedures and operator interfaces. This process melds aspects of the PRA and the more traditional deterministic evaluation methods of 10CFR50. This method has resulted in a PRA model that is structurally stable.

Comment on Acknowledgments

The flowery and effusive wording, i.e. "tireless, creative, and professional", detracts from the scholarly tone of the document and smacks of self congratulations. Other wording such as ". . . the result of the time and talents of people on the NRC staff . . ." is more appropriate.

Comments on Section 1 "Introduction"

Section 1.1 references NUREG-1335 as providing the format and content guidelines for the IPE. It should be noted that NUREG-1335 guidelines apply only to those IPEs performed according to the IPEM and not to those methods deemed "other".

Comments on Section 2 "Impact . . . on Reactor Safety"

Table 2.1 in Section 2.1.1 and Sections 2.2 and 2.3 note plant improvements which have been implemented. In these sections certain plants are specifically mentioned by name. Most of these improvements have also been implemented at SSES, including installation of a portable diesel to provide battery charging during SBO, modification to the RHRSW system and procedures for tying the fire water system into RHR/RHRSW for RPV injection and containment injection, procedure changes for avoiding loss of low pressure injection sources from inhibition of ADS during non-ATWS transients, complete revision to the ATWS procedures and others. This comment also applies to Sections 9.3.1.2 and 9.4. That is, while the Susquehanna analysis is specifically designated an "outlier", in fact most BWR "improvements" have been incorporated in the Susquehanna plant and are credited in the IPE analysis. The criticism of the SSES analysis must be balanced against the actual physical improvements in equipment, procedures, and training implemented.

Comments on Section 3 "Core Damage Frequency Perspectives"

1. Page 3-7 Add to Important Plant Improvements for Transients with loss of decay heat removal, "Operation of RWCU in the blowdown mode as an alternate decay heat removal system". This mode of RWCU operation is equivalent to feed and bleed for the BWR plant. Calculations demonstrate that operation of RWCU in this mode is capable of maintaining the containment and the RWCU equipment within design parameters.

2. Page 3-7 ATWS accidents- NEDO-32164 identifies a mode of core damage from overpower during density wave oscillations. During these oscillations the clad temperature in high power bundles may exceed rewet and may heat up beyond 1500 °K damaging the core and releasing the gap inventory. In turbine trip ATWS without MSIV isolation this represents a release that bypasses both primary and secondary containment. Core damage is prevented by prompt runback of the feed flow to the point where the RPV water level drops below the feedwater sparger. This operator action is critical for avoiding core damage during the most likely ATWS events. The draft NUREG-1560 does not address this mode of core damage, yet millions of dollars have been spent by both NRC and the industry to find ways to either prevent or mitigate it. See the NRC SER on ATWS stability.

3 Page 3-7 ATWS - "Availability of HPCS for mitigation". Should this be HPCI? BWROG EPG Rev. 4 does not allow use of HPCS for ATWS mitigation unless it is being used for boron injection.

4. Page 3-16. The statement, "The Susquehanna CDFs are lower than other BWR 3/4 plants because of optimistic modeling (relative to other plants) of success criteria, operator actions, and common cause failures." is an inaccurate representation of the Susquehanna IPE. Each area of purported optimism is addressed below.

Section 3.2.2 attributes low values of CDF at SSES to "optimistic" modeling. As discussed in volumes 3, 4, and 6 of the Susquehanna IPE, the modeling is sufficiently realistic to reveal areas for improvement in plant equipment, procedures, and training. In fact, this modeling approach has led to the incorporation of substantially all improvements identified in the NUREG-1560 draft applicable to BWR 4s with mark II containments. This modeling is the only one which includes the effects of unstable core oscillations in

ATWS, an effect which the industry and NRC have spent significant resources in terms of time and money in mitigating. It may be argued that the "pessimistic" modeling found in other analyses results in failure to identify further opportunities for accident mitigation such as saving the core in-vessel. The core damage progression used in the IPE is taken from detailed, NRC sponsored, computer codes written specifically for the BWR and supported by such data as the DF-4, CORA and X-Reactor experiments. It appears that other "success criteria" such as using TAF or a single fuel temperature as the definition of core damage are overly simplistic.

The success criteria used in the Susquehanna IPE are based on calculations performed using the NRC developed BWRSAR code for transients and LOCAs and the PP&L developed SABRE code and BWRSAR for ATWS. Additionally, the BWRSAR LOCA calculations were augmented with RELAP calculations.

Transients and LOCAs

The onset of core damage is indicated by the statement, "first fission product release", in the BWRSAR output. This output flag is based upon a correlation developed from the multi-rod bust test performed at ORNL. These test were performed to determine realistically when core damage could be expected. This feature of the BWRSAR code was used to determine when core damage occurs for the Susquehanna IPE. Success criteria in terms of equipment were developed by identifying the minimum complement of equipment necessary to avoid core damage for each type of accident sequence. The BWRSAR code was used in a number of NRC research sponsored severe accident studies, such as NUREG/CR-5565, NUREG/CR-5869 and others. The same output flag was used to identify core damage in these studies. Therefore, the success criteria for core damage are consistent with those derived by the NRC.

ATWS events

The Susquehanna IPE addressed two types of core damage for ATWS: loss of clad integrity from over power events and loss of cooling. Susquehanna may be unique in its treatment of the former type of core damage, although it is identified by GE in NEDO 32047, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability". This type of Clad damage from over power was conservatively evaluated by assuming clad damage occurs whenever the reactor enter regimes of chaotic instabilities. These regimes were identified using the PP&L developed SABRE. The NRC provided a favorable review of the SABRE results in their SER on ATWS stability issues, (Letter from D. E. Matthews, USNRC to K. P. Donovan, BWROG, Acceptance of Proposed Modifications to the BWR Emergency Procedures Guidelines TAC Nos. M89489 & M89629, 6/6/96). BWRSAR was used to evaluate core damage from loss of cooling. Therefore the success criteria for ATWS are consistent with similar NRC analyses.

Optimistic Treatment of Common Cause Failure

"Pessimistic" assumptions for common cause failure and human error, if large enough, mask improvements to plant equipment, procedures and training. Section 2.2 of Volume 6 of the Susquehanna IPE specifically addresses the IPE approach to CCF. CCF is accounted for in those systems which are redundant and for which data (or lack of it) supports the possibility of CCF. The discussion in Section 3.2.2 of the draft NUREG-1560 implies that CCF failure is not dealt with or was somehow ignored. Similar comments apply to the SSES approach for human error. Volume 6 addresses the effects of assumptions of human error (Section 2.3 of Volume 6) and concludes that "pessimistic" assumptions do more to mask safety improvements than to improve operational safety. The discussion in the draft NUREG implies that the SSES analysis ignores HRA effects out of hand or as a way of simplifying the analysis. This connotation is inaccurate.

The approach towards common cause failure is tailored to identifying and to rectifying vulnerabilities rather than bounding the incident rate of a nuclear accident. Consider the treatment of common cause failure of the diesel generators in the Susquehanna IPE. Modifications were made to the plant to eliminate design vulnerabilities that resulted in two combinations of two diesel failures that resulted in SBO given LOOP. After the modification, failure of all four diesels is required for an SBO given a LOOP. Prior to the modifications the probability of core damage from SBO was estimated to be 2.6×10^{-6}. Using typical CCF factors for diesel generators, the core damage frequency post modification is computed to be 1.0×10^{-6}. The cost of these modifications cannot be justified on the bases of this risk improvement. However statistical analysis of the actual diesel failure data demonstrates such a common cause couple between 4 diesels is improbable. In fact the analysis suggest that the common cause couple is no greater than independent failure. Evaluation of the diesel failure data since issuing the IPE further supports the improbably of such high common cause couple. Assuming a large common cause couple discourages making safety improvements to the plant and therefore is inconsistent with the purpose of the IPE which is to fix vulnerabilities.

The Susquehanna IPE was performed using an "other systematic evaluation method". This method was submitted to the NRC and found to be acceptable. The human reliability analysis methodology applied to this "other systematic evaluation method" is fully consistent with the purpose of human reliability analysis as stated in NUREG-1335, that is, to make the plant safer through human reliability analysis, and the primary goal of the IPE program as stated in the Executive Summary to NUREG-1560, which is to, "identify plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements". Rather than computing human error rates that are high relative to the equipment being operated, the HRA method used in the Susquehanna IPE was used to identify and resolve vulnerabilities from human error. A number of modifications have been made to the Susquehanna plant design, the emergency procedures and the operator training program to either eliminate these vulnerabilities or reduce their impact. This process has resulted in real safety improvement and this improvement is reflected in a vast improvement in the operators reliability at executing EOPs. These modifications include deviations to the BWROG procedures that other IPE identify as significant but haven't changed. As an example PP&L has developed procedures that eliminate many of the dominante operator errors identified in other IPEs (See recent NRC SER on ATWS stability) and removed the ADS inhibit step from the RPV control procedure.

5. Page 3-17 bulldot Susquehanna ⁻ replace the phrase, "*use of additional systems and actions*", with "*the improvements derived from modifications to equipment, procedures and training designed to eliminate or ameliorate severe accident vulnerabilities identified through its accident management process*". Additionally, the phrase, "*when supported by statistical analysis of operational data*", should be added to the last sentence which ends with the word improbable. Common cause failure data was used where its application could not be discounted by operational data. As an example, a β factor of 0.067 was used for the probability of the second diesel driven fire pump failure given failure of the first. Inappropriate use of common cause failure tends to mask vulnerabilities to severe accidents.

6. Page 3-19 DHR. see first comment in "Comments on Section 3"

7. Page 3-20, "SBO accidents are important contributors for most plants in this class. This observation is a strong function of the number of diesels on site and the assumed common cause couple. The Susquehanna plant has been modified so that 4 diesels must fail prior to losing all onsite AC power. Prior to the modifications, two combinations of two diesel failures resulted in an SBO. The diesel generator reliability data used in the Susquehanna IPE was derived from plant data. Based upon Susquehanna plant data, it is improbable that the typical common cause failure probabilities assumed in most IPEs are a realistic characteristic of the Susquehanna generators. Had typical CCF probabilities been used instead of plant data, the diesel cooling modifications discussed in the 3rd comment on Chapter 3 could not be justified based upon risk reduction arguments. Therefore when estimating the probability of common cause failure extreme caution must be applied, otherwise inappropriate conclusion may be derived.

8. Page 3-22, Operator fails to depressurize- The ADS inhibit step has been in the BWROG EPGs for over a decade.

9. Page 3-22, ... a 19% reduction..., Clearly given the uncertainty in the PRA inputs, model, etc., 19% is minutia.

10. Page 3-23 This section should address the issue of core damage from unstable operation (see NRC SER on ATWS Stability, Letter from D. E. Matthews USNRC to K. P. Donovan TAC Nos. M89489 & M89629). Additionally, modifications to procedures, equipment and training were incorporated into the Susquehanna to reduce the ATWS risk.

11. General Comment on core damage frequency calculation. The NUREG should address the criteria used to determine what constitutes core damage. Many IPEs use core uncovery while others use 2200 F. This is important since it impacts what equipment can be used to avoid core damage. As an example, at the Susquehanna plant, two CRD pumps are sufficient to prevent core damage at high pressure, but allow the water level to

drop below TAF for about 2 hours (see 16th Water Reactor Safety Meeting Feed Water Coast Down Measurement of a BWR). Such differences can have a major impact on the calculated frequency of core damage. A more subtle observation is that it has a major impact on the containment, "back ...." analysis. As an example the fraction of core damage events that occur at high pressure vs low pressure is effected by the ability of the CRD pumps to avoid core damage.

Chapter 4 Containment Performance Perspectives

1. The NUREG should address the criteria used to determine vessel failure. During the mid 80's PP&L used the MAAP code for severe accident analysis. Using this code, the core damage progression could not be terminated in vessel once the peak core node temperature exceed 3000 F. This criteria resulted in a conditional probability of vessel failure given core damage of 0.3. Later PP&L switched from MAAP to the NRC developed BWRSAR code for evaluating accident progression up to the point of vessel failure. Using the BWRSAR code, the damage progression could not be terminated in vessel once the bottom head dried out. This criteria is conservative, since the lower head debris bed can still be cooled as long as the bed temperature remains below the rewet temperature. Additionally, this criterion is supported by research performed around the world, (See papers by S. Basu & Speis, Huhtiniemi, et al, Magallon &Hohmann, and NUREG/CR-6133). This conservative criteria results in a conditional probability of vessel failure given core damage of 0.01 instead of 0.3 using MAAP. This drastic reduction is a result of two items. First, the amount of time available to the operator to restore vessel injection is increased from about 45 to 80 minutes to about 210 to 250 minutes. Therefore the probability of restoring vessel injection is substantially increased, especially in the SBO case. Second, the required injection flow rate to the RPV necessary to prevent vessel failure is substantially reduced when BWRSAR is used instead of MAAP. Using BWRSAR the injection flow requirements are less than 300 gpm and reduce with time. This requirement is well within the capability of the fire main. Using MAAP the injection flow requirement is 3800 gpm, which is greater than a single core spray pump.

This difference is illustrated using the short term low RPV pressure SBO sequence. In this example a Loss of Offsite Power has occurred with an unrecoverable failure of both HPCI and RCIC. The probability of vessel failure given core damage can be computed using recovery of equipment with the success being determined by the amount of time available. This result is presented in the following Tables.

Sensitivity on Conditional Probability of Vessel Failure Given Core Damage
MAAP vs BWRSAR/MELCOR

Accident Sequence - SBO with Failure of RCIC/HPCI

Event	MAAP	BWRSAR	MELCOR
		(minutes)	
Core Uncovery[1]	27	24	28
Core Damage[2]	42	42	65
Vessel Failure Unavoidable[3]	78	237	248

Notes:

1. Credit for feed water pump coast down was removed from the BWRSAR results since it was not included in the other calculations. Feed water pump coast down increases the time to core damage by 15 minutes.

2. Core damage is assumed to occurs for MAAP when the average node temperature reached 1500 F. Core Damage for BWRSAR occurs when the code states, 1st fission product release. Core damage was estimated for the MELCOR case by comparing the Peach Bottom MELCOR Case which did not report the time of core damage with the Lasalle MELCOR case that did in NUREG/CR-5331.

3. Vessel failure cannot be avoided in MAAP when the core debris begins relocation, a peak node temperature of approx. 3000 F. Vessel failure is assumed to be unavoidable in BWRSAR when the bottom head dries out.

Computation of the Conditional probability of Vessel Failure Given Core Damage

Equipment Recovered or Used	P(vessel Failure/Core Damage) MAAP	BWRSAR	MELCOR
1. No recovery of any equipment	1	1	1
2. Recovery of OSP per NUREG 1032	0.687318	0.342342	0.440893
3. Item 3 plus fire main injection per GL 88-20 with 10% failure rate	0.687318	0.034234	0.044089
4. Item 2 plus diesel per SSES IPE	0.577347	0.011297	0.014549

2. Page 4-18. Paragraph starting with, "The approach used in Susquehanna..." The conditional probability of containment failure given core damage is due to the following reasons:

The NRC developed BWRSAR/CONTAIN codes were used to analyze the plant response to severe accidents instead of the industry's MAAP code. As described in the general comment this provided substantially more time and significantly reduced requirements to preserve the vessel integrity. This allows for a substantial improvement in calculated containment performance.

The Susquehanna mark II design, as described on page 4-17, is perhaps the most robust containment from a severe accident view point. The floor is flat; there are no drain holes in the inner pedestal region and the procedures call for initiation of drywell sprays far in advance of vessel failure. Therefore Susquehanna is less likely to fail do to design features.

A number of modifications to the Susquehanna plant design and procedures have been made which further enhances the performance of the Susquehanna Containment. These include:

- Installing a mobile diesel generator which provides continuos supply of DC power for operation of RCIC, HPCI and the SRVs.

- Availability of compressed gas bottles that provide for control nitrogen for beyond 72 hours,

- Instruction to align the fire main for injection during SBO and to initiate a controlled depressurization to provide for RPV makeup should high pressure systems fail,

- Installation of valves which allow rapid connection of the fire main system into the RHRSW system for drywell spray application.

The tone of the discussion about SSES modeling on page 4-18 is that somehow the analysis ignores issues which are important such as containment failure modeling, human error in following procedures, etc. This is not the case. The SSES IPE models all aspects of severe accident behavior, including CCF, HRA, and containment failure, in the most realistic way known at the time of the analysis. For example, the BWR EOPs do in fact cover all aspects of the accident from on-set of the transient/LOCA through core-on-the-floor. The SSES modeling explicitly accounts for unstable core operation in ATWS. The IPE analysis deliberately strives to partition accident sequences by core damage, vessel failure, and containment failure, thereby segregating those sequences which contain the most uncertain phenomena. The approach used in the Susquehanna IPE is predicated on the idea that identification of areas for improvement may be achieved by detailed review of the accident progression, with general ideas of the timing and plant damage states. This analysis is supported by research current at the time the analysis was performed. Given that virtually every opportunity for BWR improvement noted in the draft NUREG has been discussed in the SSES IPE, it appears that this defense-in-depth approach is a viable method of performing PRA.

3. Page 4-18. 2nd paragraph from the bottom beginning with "Several Assumptions used in the Susquehanna IPE". The description of the core melt, vessel failure and corium interaction is consistent with what was used in the Susquehanna IPE. Additionally, this methodology is entirely consistent with the metallic pours the NRC used to close out the Mark I liner melt through issue as described in NUREG/CR-5423. Additionally, this treatment of core damage progression is entirely consistent with the treatment in NUREG/CR-5623, BWR Mark II Ex-Vessel Corium Interaction Analysis. This treatment is a direct result of using NRC developed codes rather than the MAAP code.

Page 4-20 Paragraph - Condensations loads peak at about 165 F in the pool. At higher temperatures the collapse rate of the steam bubbles is too slow to impart large loads, see SRI international research report, "A Summary of Test Results on Performance of Quenchers at High Suppression Pool Temperatures", February 1980.

Comments on Section 5, Human Action Perspectives:

Section 5.1 indicates that despite different HRA approaches, a relatively small set of important operator actions was discovered. It should be pointed out that all are actions for which no automatic back-up exists, and that this alone is a good criterion for selection of "important" human actions.

Consistent with comments above reflecting the state of the art of PSA in general, it is not surprising that HRA results differ among plants/studies. A number of BWROG member utilities are now undertaking a structured peer review process on a trial basis. It was discovered during these reviews that while HEP values may appear to vary widely, upon further investigation it was found that such variations reflect more on the definitions used for the actions than variation in the actions themselves. When consistent definitions are applied, consistent HEPs result.

The variability of HEP values noted in Figure 5.1 should also be credited to the variability of EOPs themselves. For example, BWR EOPs at some sites do not require the inhibition and later un-inhibition of ADS during transients. Removal of these steps from EOPs removes these HEPs.

The discussion of Section 5.3.1 implies that all HRA approaches used to date are deficient. This discussion is consistent with that in NUREG-6350 "A Technique for Human Error Analysis ATHENA". This sentiment is also reflected in the Executive Summary of NUREG-1560 (page xviii) which states that "An HEP is only valid to the extent that a correct and thorough application of HRA principles has occurred." However the Summary also states that there is little evidence that the HRA quantification method has a major impact on the PSA results. HRA is thus portrayed as a "wild card" which nonetheless seems to have little effect on results. The arguments put forth in the draft NUREG-1560 suggests that minimal time be spent on future improvement/refinement of HRA because the time spent won't change the results very much. This is also the only way such widely varying HRA results can be described as "reasonably similar" in Section 5.4.

The discussion of HEPs indicates that simulator exercises may be used to refine HEP estimates. Susquehanna makes extensive use of the results of simulator exercises. It should be noted on page 5-16 that although SSES HEPs are low, these values are supported by simulator exercise results.

Page 5-16 The Susquehanna plant is an outlier on the depressurization step. The Susquehanna RPV Control Procedures do not instruct the operator to inhibit ADS.

Chapter 7 Additional IPE Perspectives.

A CCFP of greater than 0.1 for 26 BWRs is possibly a consequence of using the MAAP code instead of BWRSAR or MELCOR. Using BWRSAR or MELCOR allows credit for fire water when terminating the damage progression in-vessel. Terminating the damage in vessel prevents containment challenges from all the ex-vessel phenomena. Using MAAP requires quenching the damage progression prior to debris relocation. Therefore the probability of vessel failure is much higher. Therefore additional containment challenges must be considered. See comment Chapter 4 comment 1 for additional discussion.

Vol. II

1. Page 9-35 Add mobile generator to supply DC system. This modification reduces the chance of high pressure melt ejection during SBO caused by loss of DC control power to the SRVs.

2. Page 9-40. Alternate Water Supplies - Susquehanna has credited the use of fire water for drywell sprays. Additionally, valves have been added to facilitate connection of the fire main system to the RHRSW line.

Page 10-14 The Susquehanna Plant has deviated from the BWROG and does not Inhibit ADS. Operation of both CRD pumps is sufficient to avoid core damage in short term sequences at the Susquehanna Plant.

Page 10-16 Susquehanna has 5 diesel generators. One of the 4 can energize any of the 4 ESS busses

Page 10-30 Containment Heat Removal. - The staff should peruse the blowdown mode of RWCU as an effective mode of DHR. Using RWCU in this manner no design limits are violated. Additionally unlike venting no containment gasses are released. This method was submitted to NRR during the late 80's. They had ANL review our calculations and concurred with its use.

NEI REVIEW OF DRAFT NUREG-1560
- PRELIMINARY COMMENTS & OBSERVATIONS -

OVERALL INDUSTRY RESPONSE

1. NUREG-1560 contains a wealth of data, insights and conclusions from the 75 IPEs submitted in response to Generic Letter 88-20. It will undoubtedly serve as a benchmark for US nuclear safety and, therefore, warrants careful review and comment by the industry.

2. Much of its contents are devoted to the analysis of IPE data, the characterization of various plant types and extraction of applicable insights for both specific plants and NRC-defined plant type groups. Individual utilities and owners' groups should carefully review these characterizations and provide specific comments to the NRC.

3. Several chapters of the document are devoted to higher level insights and conclusions which could have a direct impact on the NRC's future policy making. This review focused on these chapters and is intended to be input into a higher level comment package which may be organized by NEI through the Risk-Based Applications Task Force (RBATF). The majority of the following comments are oriented toward the RBATF comments.

4. Finally, the overall document (both volumes) is over 800 pages long. Providing a complete technical response by March 15th will be very difficult for the industry. In addition, it seems that dialogue with the staff during a workshop could help focus these comments and benefit both the staff and the industry. An additional round of comments will be entertained after the workshop, due date May 9th.

EXECUTIVE SUMMARY IN GENERAL

1. The IPE process succeeded in incorporating risk insights and PSA into many activities. Consequently, utilities have a greater appreciation for risk sensitivities and the difference between the traditional deterministic safety bases and the risk insights gained from PSA analyses. This has lead to a number of proactive programs being undertaken by utilities. The industry should take every opportunity to remind the NRC and the public that over 500 plant safety enhancements were

identified in the IPEs and that over 225 were implemented before submittal of the IPEs (pg. xii).

2. The overall conclusions of this report with respect to specific numerical estimates should be softened. The industry has also done a lot since the submittal of the IPEs to enhance safety. As the industry positions itself to engage in a risk-informed regulatory environment, utilities have sought to reduce core damage frequencies in accordance with the EPRI PSA Applications Guide (EPRI NP-105396) screening criteria. These criteria encourage plants with high CDF and large early release frequencies (LERF) to reduce these risks in order to support application of PSA in other regulatory arenas. This is indicated in the data the WOG presented showing that many of the previously high CDF plants have brought their CDF below 1E-4/yr. This point is important for three reasons:

   - computed CDFs are continuing to go down without further regulatory action
   - the results of the IPEs are not necessarily indicative of the current risk levels
   - conclusions about the ranges of results, risk to the public and even dominant contributors based on the IPEs may not be indicative of current results.

1. Conditional containment failure probability (CCFP) continues to be used by the NRC as a safety indicator (pg. xv). The industry has repeatedly demonstrated the inadequacy of this figure of merit (in the PSA Applications Guide and at the ACRS PRA Subcommittee Meeting on 8/7/96). This report provides further evidence that this parameter is not useful (see comment below on safety goals). We should be sure to challenge this as an appropriate figure of merit.

2. The staff states that "before an IPE/PRA can be used beyond its original purpose (GL 88-20), the quality of the IPE/PRA will need to meet the standards established for a specific application… (pg. xviii)." This is not entirely consistent with past NEI positions, but is generally consistent with activities currently underway (e.g., EPRI ESAF, BWROG certification, CEOG cross comparison, etc.). This is an area where further dialogue with the NRC would be useful. (See comments on attributes of quality PSA)

3. The NRC found the "quality" of Level 1 portions of the PSAs to be generally good [robust] (pg. xix). However, the NRC specifically raises issues about the following areas:

   - plant-specific data
   - common cause failure data
   - human reliability data (this was particularly highlighted)

   These issues are not new. Before the IPEs were initiated one could have identified these as key areas of uncertainty. The issue with plant specific data seems to involve two aspects: (1) limited use of plant experience by some plants and (2) statistical techniques used in dealing with small data samples. This issue can be addressed by throwing money at the problem at each plant, but there is a good argument that plant specific data has a secondary (or tertiary) impact on the overall results, except for a few key systems/trains/components (see EPRI Data Rule comments submitted by NEI). There are competing arguments that the use of generic data helps plants in addressing failures which have not occurred at their plant, but could.

   The issue with common cause failures seems to have three aspects: (1) while virtually all IPEs reference NUREG/CR-4780 as their method of treating common cause failures, the implementation varies considerably (one should also recognized that the level of consistency between NUREG/CR-4780 and NUREG-1150 is also questionable in some experts minds), (2) the scope of components treated in many of the IPEs (and NUREG-1150) is relatively limited, and (3) INEL, under contract with the AEOD of the NRC, has recently completed an update of the NUREG/CR-4780 common cause databases to include significantly more failures and component types and the NRC is now viewing this new, essentially unknown, database and associated software package as "the" proper common cause approach.

   The human reliability issues are much less well defined. The NRC's specific technical issues are not well characterized in NUREG-1560 and no solutions or acceptable approaches are identified. Some members of the staff seem to be pointing to the NRC's research HRA program methodology, ATHENA, as the solution, but this method is still under development and has yet to proven. In addition, the staff has been inconsistent in their SERs with respect to HRA. One recent SER on a PWR IPE specifically cited a previously accepted HRA method as

being adequate for IPE purposes but inadequate for applications purposes, despite the fact that the PSA was already under review (without comment) by NRR as a part of one of the pilot risk-informed regulation applications.
x
The truth is that both common cause and human reliability data are sparse, difficult to interpret and arguable until the end of time. Further research will undoubtedly be undertaken by the NRC and industry. However, progress on risk-informed regulation should not be held up pending this research. The PSA application process must be robust enough to allow progress without resolution of each and every technical issue.

1. The "quality" of the Level 2 portions of the PSAs is called into question in a number of places. Especially those "which relied heavily on either the use of the MAAP code or the use of a set of industry position papers (pg. xix)." First of all, the NRC accepted MAAP 3.0b for use in the IPEs and essentially every IPE used MAAP in some capacity. While there have been some mis-applications of MAAP (i.e., those not complying the EPRI guidelines for application), any implication that the MAAP code is inadequate is wrong. Furthermore, MAAP 4.0 is now available and has received a much broader endorsement from the NRC.

   A more problematic item involves the utilities which did not properly apply MAAP and/or relied on the so-called industry position papers. There are a number of such utilities and the NRC position may have technical merit in those cases, with respect to applications. Nevertheless, the industry should be careful in "globally" accepting <u>or</u> rejecting this finding. The PSA certification process (or equivalent) must address this issue in a manner which is both technically defensible and feasible.

COMMENTS ON OVERALL CONCLUSIONS AND OBSERVATIONS

Comments listed in order of the bullets on Page xx:

1. There is no question that the IPE program has had a substantial, positive impact on how utilities have looked at safety and has significantly enhanced most utilities understanding of PRA and severe accidents. However, the slow rate at which progress is being made on risk-informed regulation jeopardizes the industry's ability to maintain that expertise. Without promise of further usefulness, PSA will not retain its current standing and utility support for expending resources on PSA maintenance and update will not continue.

2. Staff follow up activities in the areas of:

   - plant improvements (this should be expected)
   - containment performance improvements not implemented or addressed (this should be expected)
   - plants with relatively high CDF or CCFP (we should try to influence this).

   The staff follow up on high CDF/CCFP plants has two problems. First, the IPEs were not necessarily a search for a true CDF/CCFP, they were a search for vulnerabilities. In fact, prior to initiating the IPE process the NRC repeatedly emphasized that the absolute values of the IPEs were not of interest, but rather the relative importance of sequences, systems and human actions. As such, in the context of the IPE, if conservatisms were included, it was probably OK. (The degree to which, and the manner in which, recovery was applied may be one such example.) Consequently, some studies that were submitted as IPEs contained conservatisms which skewed the results. Watts Bar and Sequoyah are good examples of this. They were both submitted at CDFs greater than 1E-4/yr (3.3E-4/yr and 1.7E-4/yr, respectively) and then subsequently updated with better plant specific analyses and both were found to have CDFs near 4E-5/yr. Furthermore, NUREG-1560 identified 15 PWRs with CDFs greater than 1E-4/yr and the vast majority (>80%) of BWRs and PWRs as having CCFPs greater than 0.1.

   It must be emphasized that the purpose the IPE was to identify vulnerabilities. Generic Letter 88-20 did not define "vulnerability." However, it now seems that the NRC has, after the fact, identified this CDF and CCFP criteria as their definition and intends to use it. There is no technical basis for these criteria as vulnerability definitions, in fact they were originally defined as "goals," not criteria. This issue should be addressed strongly by the industry.

The second problem with these CDF/CCFP criteria is that neither has any established technical connection to the QHOs of the Safety Goal. As described by the staff in Chapter 7 of NUREG-1560, CCFP has no correlation with the QHOs and CDF has only a limited correlation as a precursor to releases. In Chapter 7, the staff had to compute the frequency of early releases containing greater than 0.03 of the Cs, I or Te inventory (a figure of merit very comparable to the industry's LERF) in order to even do a comparison to the QHOs. Not surprisingly, based on NUREG-1150, the industry's definition of LERF actually has a better correlation to the QHOs than CDF or CCFP.

1. The staff's expectation that further review may be necessary for application of IPEs in risk-informed regulation is consistent with EPRI PSA Application Guide and NEI's draft process for PSA certification. The PSA Applications Guide outlines numerous steps that should be taken to assure that the PSA is applicable to the change being evaluated in the application. Furthermore, industry should be responsible for establishing and assuring the appropriate quality of its PSAs.

2. The USI and GSIs identified for possible resolution should be monitored by industry groups. In particular, GSI-23 (RCP seal failures) seems to be cropping up again. The IPEs do provide some ammunition to the NRC with respect to the risks due to RCP seal failures due to loss of cooling. However, what may not be obvious is whether some of those sequences may have gone to core damage even if the seals had remained intact. For example, a dominant contributor to RCP seal LOCA core damage is often loss of all essential service water. However, in many plants, essential service water usually also provides cooling to components, rooms and buildings that would be necessary to perform a normal shutdown, even without an RCP seal failure. Therefore, "perfect" RCP seals might somewhat reduce the risk of loss of service water, but not eliminate that risk. This will be hard to determine from a simple review of IPE documentation.

3. Research activities identified in Section 8.5 are no surprise:

   - Human Performance

- Risks Of Other Modes Of Plant Operation
- BWR Containment Venting
- Ice Condenser Containment Failures
- Severe Accident Phenomena

1. Conclusions based on using the IPE results for comparisons to the QHOs of the safety goal must be carefully qualified. As stated above, the purpose of the IPEs was not to define absolute risk levels, but rather to identify plant severe accident vulnerabilities. Consequently, the safety goal computations performed by the staff (described in Chapter 7) are not an adequate technical basis on which such conclusions can be drawn. Several of the plants identified in Chapter 7 as potentially approaching the early fatality QHO have subsequently updated their PSAs with significant reductions in CDF and LERF (including Browns Ferry, Beaver Valley and Palo Verde). This is a case where the NRC has tried to apply the IPEs in a manner not consistent with their original intent…something they have gone to great pains to admonish the industry for in risk-informed regulatory applications.

2. Clearly, the industry agrees that the IPEs/PSAs can be utilized to support a variety of safety applications and that the schedule for such should be expedited.

3. The term "standardization" can mean a lot of different things to different people. The industry endorses efforts to improve IPEs to support applications through certification of the inputs, models and results. However, if the NRC's "standardization" involves an across the board incorporation of all the "quality" attributes identified in Chapter 6 before risk-informed regulation can proceed, then the industry has significant cause for concern. Specifically, the industry does not endorse the development of mandatory "standards" (e.g., ANSI Standards) for PSA. Good practices, technical positions, accepted methods, etc. are more consistent with industry activities.

   Quality attributes of Chapter 6 are generally consistent with BWROG PSA certification criteria. This is good. However, three potentially significant issues merit more dialogue with respect to these attributes. First, it is not clear if these attributes are minimum requirements for applications or a statement of the state of the technology for PSA. The report seems to be somewhat contradictory on this point. Secondly, if this is a statement of the minimum requirements, it would be easy to show that the NRC's flagship study, NUREG-1150, does not even meet all of these attributes. Finally, unless these are a statement of the state of the art, a number of specific technical items need to be discussed. Nearly every PSA in the industry would require re-work to meet all of these attributes.

The Level 2 quality attributes, in particular, are more comprehensive than required for most applications.

ITEMS IDENTIFIED IN A CURSORY REVIEW OF OTHER SECTIONS

1. The discussion of the Maintenance Rule says it is acceptable to use the IPEs to determine risk significant systems. However, this is not compatible with the findings about the usefulness of the IPEs for risk-informed regulation. Likewise, the NRC implies that for inspection purposes the IPEs are adequate for them to target areas for plant-specific inspection (pg. 8-14,15), but in the Executive Summary the PSAs are identified as only adequate to identify dominant accident sequences types and not their relative importance. This seems inconsistent. Furthermore, the NRC seems to be attempting to use PSA information in a selective manner, where it serves their purposes. PSAs use in design basis inspection activities seems inconsistent with the NRCs positions on risk relative to licensing basis (e.g., PSA is outside consideration in the licensing basis).

2. Some of the potential new generic safety issues merit investigation by owners groups. (pg. 8-13,14):

    - RCP Seal LOCAs (PWRs)
    - Inhibiting of ADS in BWRs (BWRs)
    - Impact of Accident Environments (PWR & BWR)
    - Blowdown of BWRs Into Hot Suppression Pool (BWR)
    - Induced SGTRs (PWRs)
    - Induced Isolation Condenser Tube Ruptures (BWRs)
    - Prevention of BWR Mark I Shell Melt-through (BWRs)

1. The report utilizes at least five different figures of merit in characterizing containment performance. It is never clear which figure is the most appropriate, or why. The figures used in NUREG-1560 include:

- total conditional containment failure probability (the subsidiary safety goal) is cited in the report, but is never reported or displayed. This figure of merit implies that all containment failures are important.
- conditional probability of various containment release types: bypass, early failure and late failure (pg. 4-5 & 4-27). These figures of merit simply sub-divides the total, but imply that early failures and bypasses may have some significance.
- frequency of bypass and early release (pg. 7-2). This figure of merit implies that early failures and bypasses are most important.
- conditional probability of "significant early release" (pg. 4-45) (defined as a release containing > 0.1 of core Cs, I, Te inventory). This figure of merit implies that only some early failures and bypasses are important.
- frequency of releases with the potential to cause early fatalities (pg. 7-5) (defined as a release containing > 0.03 of core Cs, I, Te inventory). This figure of merit implies that fewer of the early failures and bypasses are important.

x

The industry has long pointed out the weaknesses of conditional probabilities of containment failure, but the staff continues to focus on those figures. For BWRs where the coupling of core protection with containment response is intimate, CCFPs can be very misleading. As noted in the executive summary (pg. xix), BWRs tend to have higher CCFPs than PWRs, but lower CDFs, resulting in comparable release frequencies. The higher CCFPs for BWRs are not necessarily an indication of "weaker" containments, but are due in no small part to the coupling of containment systems and core damage prevention systems like (RHR). It is therefore better to focus on the <u>frequency</u> of containment failures (especially large early ones).

1. A casual review of Section 14 of Part 4 identified that the NRC believes that peer review is an "essential part of a quality PRA." However, they also state that "it is not appropriate for other utilities to perform the peer review (pg. 14-66)." This is not consistent with industry thinking on this subject.