WESTINGHOUSE OWNERS GROUP
WOG
ESTABLISHED 1979

Domestic Utilities
American Electric Power
Carolina Power & Light
Commonwealth Edison
Consolidated Edison
Duquesne Light
Duke Power
Georgia Power
Florida Power & Light
Houston Lighting & Power
New York Power Authority
Northeast Utilities
Northern States Power
Pacific Gas & Electric
Public Service Electric & Gas
Rochester Gas & Electric
South Carolina Electric & Gas
Southern Nuclear
Tennessee Valley Authority
TU Electric
Union Electric
Virginia Power
Wisconsin Electric Power
Wisconsin Public Service
Wolf Creek Nuclear

International Utilities
Electrabel
Kansai Electric Power
Korea Electric Power
Nuclear Electric plc
Nuklearna Elektrana
Spanish Utilities
Taiwan Power
Vattenfall

OG-97-031

Project Number 694

March 25, 1997

Ms. Mary Drouin
U.S. Nuclear Regulatory Commission
MS T10E50
Washington, DC 20555

Subject: Westinghouse Owners Group
Westinghouse Owners Group Comments on NUREG-1560 "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance" (MUHP6029)

Attached for your information and use are the comments from a review of the draft version of NUREG-1560 which were compiled by the Westinghouse Owners Group Risk-Based Technology Working Group (RBTWG). Comments also include those made by Westinghouse.

If you have any questions regarding this information, please contact Jerry Andre' (Westinghouse) at 412-374-4723.

Very truly yours,

Louis F. Liberatori, Jr.
Vice-Chairman
Westinghouse Owners Group

attachment

cc: Risk-Based Technology Working Group Representatives (1L,1A)
S.M. DiTommaso - WEC 5-16 E (1L, 1A)
Systems and Equipment Engineering Subcommittee Representatives (1L,1A)
Primary Representatives (1L,1A)
Steering Committee (1L,1A)
C.M. Craig, USNRC (1L, 1A)

9700031.DOC

Westinghouse Owners Group Comments on NUREG-1560
"Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance"
Summary Report
Draft Report for Comment

GENERAL COMMENTS

1: The Individual Plant Examination (IPE) process (together with certain pre-IPE studies by some licensees) has resulted in the identification and recognition of the important severe accident sequences and improvements in the plant capability to deal with severe accidents across the board in US nuclear power plants. As such, the draft NUREG correctly recognizes that the IPE program resulted in improving the overall safety of nuclear power plants.

2: The report presents useful information on the estimated core damage frequencies of various types of accidents and the range of core damage frequencies for the various types of plants. It also presents important conclusions on the containment capabilities for severe accidents. Our overall impression of the information presented in the report is that the staff has put a very good effort in assimilating and synthesizing the many pieces of information from the IPEs and presenting the key results and conclusions with respect to the severe accident risk for US plants in an informative manner. But, it needs to be kept in mind that this report only provides information on the IPEs which may not be representative of plants anymore (see Comment 5).

3: Concerning any follow-up regulatory activities, it's suggested that the investigation and regulatory considerations not be limited just to the high core damage frequency (CDF) or conditional containment failure probability (CCFP) issues. Areas where the risk impact is small and the safety benefit is not appreciable should also be investigated for reduced regulatory burden, similar to the improvement in the Appendix J ground rules recently promulgated. In this regard, it is pointed out that at this time there are a number of resource intensive regulatory issues the NRC and the licensees are investigating, NRC GL-96-01 and GL-96-06, accuracy and availability of design basis information, process for performing 10 CFR 50.59 evaluations, etc. It is suggested that the NRC utilize the results, conclusions, and insights from the IPE program and the risk-based decision making to ensure that the response to these issues is properly balanced with respect to nuclear safety.

Through the IPE and related processes, understanding of the qualitative and quantitative probabilities of accidents and failure modes of equipment important to safety has substantially improved from that of the 1960s, when 10 CFR 50.59 was implemented. We suggest that this improved understanding be utilized in defining a threshold of "increase in probability" for which NRC review is needed, as contemplated in 10 CFR 50.59. This will enable focusing the licensee and NRC effort on the more risk and safety significant issues and areas of plant operation.

4: An important additional conclusion from the NUREG is that there is tremendous variability in all aspects of the studies. That variability makes it difficult to draw conclusions and it may mask important insights. Effective PSA usage may require reducing this variability. The large range of plant improvements identified, for example, may simply be due to the extreme

variability in IPE result. The following are examples, in the NUREG, of areas of significant "diversity" or variance among plants:

A. Criteria used define a vulnerability
B. Containment performance
C. Which operator actions are "important" (e.g., Pg. xvii, "Only a few specific human actions are consistently important for either BWRs or PWRs and reported in the IPEs."; and pg. xviii, "The results from this examination indicated that some of the variability in the HEP values may have been an artifact of the way in which HRA methods are applied.")
D. On pg 23, the NUREG notes, "The ATWS results are affected more by modeling assumptions than by plant-specific design features."
E. On pg 5-17, the NUREG notes, "Neither BWR nor PWR submittals show a broad consistency in terms of which human actions are found to be important."

5: The information provided in the IPE submittal reports is no longer representative of plants. Since the time when the IPEs were completed, there have been significant changes to plant designs, operator responses to events, modeling approaches and assumptions, and data. Changes directed at providing more representative PRA models of the plants have had a significant impact on dominant CDF sequences. The impact of these changes and improvements need to be considered prior to finalizing any conclusions or recommendations. The NRC should refrain from making conclusions or recommendations based on this "snapshot" analysis.

6: The results are presented by reactor and containment type and NSSS. It would be valuable for the results to also look at the architect/engineer (A/E) and/or builder and also vintage of plant to see if the variations can be explained with the NSSS category. As we have seen in our comparisons, this is an important determining factor. This is especially true for station blackout, auxiliary feedwater, and support system impacts on reactor safety. This is important because of the generic insights derived in this report and the generic implications that are made.

7: In grouping plants on the basis of "high" CDF or CCFP, the criteria as to what and why constitute "high" should be formulated carefully and objectively, considering the precision in the estimates of these probabilistic events. For example, is a plant with an estimated CDF of 1.04E-04 less safe than a plant with an estimated CDF of 0.96E-04?

8: The extra efforts implied in Chapter 15 of NUREG-1560 may make the movement to risk-informed regulation cost-prohibitive for many utilities.

EXECUTIVE SUMMARY

1 (Table E.2): The definition of early failure and bypass in this table appear to correlate to the industry definition of LERF. Some agreement between the NRC and industry should be reached on a consistent definition.

2 (Page xix): Under Additional IPE Perspectives, last sentence of first paragraph, "However, based on extrapolating the NUREG-1150 results, a few plants may approach the early fatality health objective." This is also discussed on page xx under overall conclusions/observations. Is this comment directed at BWR or PWR plants or both? In addition, inferences that a few plants may approach the early fatality health objective based on a comparison of the IPE and

NUREG-1150 results may not be valid and care should be taken in doing so, especially in drawing the previously noted conclusion.

3 (Page xx): The word "standardization" should be defined. This could have implications as to how the industry approaches the continued use of PSA.

CHAPTER 1

1 (Page 1-3): Second paragraph discussed that the perspectives in the report are based on the original analyses and that licensees have updated their PRAs to reflect plant changes and comments from the NRC staff and that these changes are not reflected in the report. This should be reiterated in the executive summary since this is an important point.

CHAPTER 2

1: Most of the vulnerabilities identified for the PWRs are not based on NSSS, but on the A/E designed systems (HVAC, electric power, compressed air, etc.; see General Comment 6). Page 2-15, "Other PWR improvements, particularly for Westinghouse plants" implies an issue with Westinghouse, but this is A/E driven.

2 (Page 2-5): The vulnerability related to an external seismic-induced SBO is not part of the IPE review (seismic is outside the scope of this report) and should be removed from the report. Also discussed in Table 2.2 on page 2-11.

3 (Page 2-10/Table 2.2): Under Westinghouse 4-loop PWR, it states in vulnerability description that SBO and ISLOCA are "(major contributor to public risk)". This should be removed from the table since it is not identified for any of the other summaries (BWR or PWR).

4 (Page 2-9/Table 2.2): Under the Pressurizer PORV block valve alignment vulnerability description, the licensee approach does not seem to match (be appropriate) the vulnerability description. Something may be missing.

CHAPTER 3

1 (Chapter 3.3): Categorizing Westinghouse plants according to the number of loops and drawing conclusions between CDF and number of loops is inappropriate. The number of loops has very little to do with the ability of a plant to mitigate events or the probability of event occurrences. For station blackout events it is more important to consider the site location, number of units per site, and number of and reliability of onsite power sources, in addition to modeling of the event with regard to power recovery and RCP seal LOCAs. For comparison of LOCA events, it is more important to consider the design of the ECCS system, the design of the systems that support the ECCS, and ECCS recirculation requirements (manual, semiautomatic, automatic, time available for operator action). Again, these differences are not a function of the number of loops. There are several small groups of plants that have similar ECCS system designs that could possible be compared, but there is no mention of this.

2 (Chapter 3.3): The choice of success criteria has also a major impact on the variability of the CDF results in a given category of plants. This is not mentioned in the NUREG (maybe there is a good reason why it is not mentioned). But some utilities working with smaller PRA vendors had more stringent (conservative) success criteria than others who worked with

reactor vendors and had access to less conservative success criteria. Also some larger utilities had the resources to seek basis for less conservative success criteria at a cost, whereas other utilities may have used less expensive method of "off the shelf" success criteria. A good example of developing the basis for less conservative success criteria can be seen in the Zion CDF.

3 (Chapter 3.3): Westinghouse 2-Loop Perspectives: Switchover to recirculation (page 3-59) and general comment on Operator Actions

Even in the same so called "groups" of plants, there are considerable fine differences that justify the differences in CDF. The NUREG states that "Point Beach licensee based the IPE on a pessimistic value for operator failure to perform the switchover". A comparison of equipment layout and emergency procedure steps was made between Kewaunee and Point Beach during the IPE model construction. The differences were significant and justified higher human error probabilities for some actions for Point Beach than Kewaunee. Such fine differences may be glossed over by a high level review, but actually are crucial for understanding the plant. As a specific example, switchover to recirculation at Point Beach requires opening manual valves locally while switchover for Kewaunee and Prairie Island is done from the control room via motor-operated valves. Such implementation differences justifies differences in HEPs.

4 (Chapter 3.3): It is not clear where special initiators fit into the CDF information reported. Generally loss of component cooling water and loss of service water can be important contributors to CDF due to the potential for a RCP seal LOCA. It would be advantageous to report the transient results in terms of CDF due to loss of decay heat removal and CDF due to consequential LOCAs.

5 (Chapter 3.3): Care must be taken when comparing CDF from transient events and from LOCA events. The various IPE models approach the modeling of consequential LOCAs (RCP seal LOCAs, stuck open PORVs or safety valves) differently. Sometime the CDF from these events are reported in the transient CDF and sometimes in the small or medium LOCA CDF. It needs to be clearly stated how this is handled in NUREG-1560.

6 (Chapters 3.3 and 11.3): NUREG-1560 needs to further consider statements on the Westinghouse RCP seal LOCA model. The Westinghouse RCP seal LOCA model appears to be more conservative than other models used in IPEs. A critical parameter in all RCP seal LOCA models is the probability of core uncovery (and core damage) occurring within the first hour. This value from the Westinghouse RCP seal LOCA model is 0.0283. Other Westinghouse IPE models that used other RCP seal LOCA models go as long as 90 minutes before considering RCP seal LOCAs. This is particularly important for SBO events, loss of service water or component cooling water events, and transient events with failures of service water or component cooling water. Please address this issue.

7 (Chapter 3.3): The discussions on LOCAs should be directed at the ability of plants to mitigate small LOCAs. Overall, large LOCAs are not significant contributors to CDF.

8 (Page 3.1): Transient induced LOCAs do not defeat heat removal through the secondary side as stated. Injection is required to make up for RCS inventory lost through the LOCA (RCP seal or stuck open safety or relief valve). Decay heat removal will still be required, unless sufficient heat is removed through the LOCA, which can be done via the secondary side by auxiliary feedwater.

9 (Page 3-35/Figure 3.9): A basis for the key perspective that "plants with better feed and bleed capability generally have lower CDFs" should be provided. There are many other plant design features and modeling methods that have a greater impact on CDF.

10 (Page 3-68): The statement about Zion CDF being lower because of optimistic success criteria, common cause, and human error modeling is a broad brush statement which is misleading. The Zion HRA was redone at the insistence of the NRC by a different consultant and the plant CDF was barely affected. Also ComEd developed the less conservative success criteria bases at a considerable cost in a programmatic manner; this bold attempt should be applauded rather than put down by saying it is optimistic. It is worse to encourage the use conservative success criteria which masks the real risk contributors.

CHAPTER 4

1 (Page 4-27): It is curious to note that, "The low early structural failures for ice condensers relative to the other PWRs appear to be driven more by analysis assumptions than by plant features." The CCFPs (conditional containment failure probabilities) for early failure are smaller for ice condenser containments that for large dry's, which is counter to any previous industry understanding of containment behavior. This discrepancy merits more investigation.

CHAPTER 5

1 (Page 5-1): The page note, "Of particular concern is the degree of variability in the quantification of similar human actions across different plants." It is agreed that this is a concern, and that the subject requires more investigation.

2 (Page 5-8): This page discusses small LOCA's and describes how some plants (PWR's) with large RWST's do not require switchover to recirculation cooling. All of these plants have NRC-reviewed EOP's, whose primary directive for small LOCA is to cooldown and depressurize, and not use cold-leg recirc. The NUREG should mention this inconsistency between EOP direction and the modeling of cold leg recirc. in the IPE's.

3 (Page 5-12): This page states that plants beside Maine Yankee cannot use Maine Yankee's method to deal with an SGTR because no other plants have reactor coolant loop stop valves. That is not entirely correct. Many plants have them (including Zion, Byron, and Braidwood). But using loop stops to respond to an SGTR was removed from the generic WOG procedures. The option was considered, and rejected, by the Westinghouse Owners Group.

4 (Page 5-15): This page claims that, regarding human error probabilities, "several either failed to consider context or dependency at all. . ." Plants known to us, which received that criticism in their SER's, have since submitted revised HRA's. Therefore, it is important to consider the updated IPE submittals.

CHAPTER 6

1: The PRA quality standards identified in Sections 6 and 14 belong in the Standard Review Plans and Reg. Guides, not in the NUREG. Otherwise there may be conflicting information in the different documents.

2 (Page 6-1): Chapter 6 discusses the authors' opinion of the features of a high-quality PSA. This includes such potentially costly and somewhat controversial features as full propagation of uncertainty through the CDF, containment performance, and fission product models. Also included is a requirement for consideration of ". . .the full range of views in the technical community" regarding phenomena. The ideas presented in this NUREG, for what constitutes a good PSA, should be clearly stated as being one set of ideas, rather than foregone conclusions. The essential features required for an effective and practical PSA should be the subject of continuing discussion with the industry.

3 (Page 6-12): The following sentence appears to state that no employees of any utility should serve as reviewers for any PSA: "Therefore, individuals who performed the PRA and other utility personnel are excluded from the peer review team." Their plant knowledge and PSA knowledge would improve the quality of any review team. Experienced PSA analysts working for utilities other than the one that performed the PSA would be excellent independent reviewers.

CHAPTER 8

1. (Page 8-14): The statement "In some cases, such as drywell shell melt-through in BWR Mark Is and direct containment heating in PWR containments agreement in the expert community has only been reached (i.e. since completion of the IPE analysis)." This discussion should be expanded to explain what agreement has been reached, and references should be cited.

CHAPTER 9

1: Tables and text in chapter 9, "McQuire" should be "McGuire."

2 (Pages 9-6 and 9-7): Tables 9.4 and 9.5 have some overlap with plants (Quad Cities, Brunswick, Robinson, Harris, Braidwood, Byron, Wolf Creek, and Zion) identified (plants appear in both tables). How can these plants fall in both categories?

3 (Page 9-14): Table 9.7, the line that states "same as Beaver Valley 1" in both vulnerability description and licensee approach. What is the "same" (all the insights or what?)?

4 (Page 9-16): The discussion talks about "Summer", but vulnerabilities for this plant are not identified in Table 9.7. How can this be?

5 (Pages 9-18 and 9-19): The column "miscellaneous" on Tables 9.8 and 9.9 can vary widely and should be explained in more detail as to examples in that category or it should be removed entirely.

6 (Page 9-26): Table 9.11, listing improvement implementation by licensees as of the date of IPE submittal, is misleading because many plant changes have occurred since the initial IPE submittals.

CHAPTER 10

1 (Page 10-19): "PCS" is BWR jargon. On PWRs, this is usually called the "secondary side" or "secondary system". We recommend a change to the terminology in the 3rd paragraph on this page.

CHAPTER 11

1 (Chapter 11.3): It is not clear from the information presented that the Westinghouse RCP seal LOCA model provides a lower contribution to CDF than IPEs that use other RCP seal LOCA models. Since this is very important to a good number of plants, it is recommended that NUREG-1560 provide a detailed comparison of the two approaches.
As previously noted, one of the dominating factors in the seal LOCA model is the probability of core uncovery (core damage) occurring within the first hour. IPEs using the Westinghouse RCP seal LOCA model typically use 0.0283 and IPEs with other models use 0.0. From this it appears that the Westinghouse RCP seal LOCA model is more conservative.

2 (Chapter 11.3): A discussion on how the component failure rates and common cause failure rates impact the results is missing. This could be particularly important for assessing the importance of SBO since the reliability of onsite emergency AC power is critical.

3 (Page 11-104): In Table 11-33, Summer should be listed as a single unit site.

CHAPTER 14

1: The PRA quality standards identified in Sections 6 and 14 belong in the SRP and Reg. Guides, not in the NUREG. Otherwise there may be conflicting information in the different documents.

2: (Page 14-20): The report states that "Using lower generic common-cause values than those shown in the AEOD report or eliminating a common cause treated in the AEOD report are discouraged and are generally deemed inappropriate." There is no basis provided for stating that the AEOD report is preferable to any other source of common cause data. The industry never commented on the validity of the methods or interpretation of results.

CHAPTER 15

1: The conclusions from this chapter should be reiterated and not lost. This chapter concludes that for a majority of the PRAs, the PRAs meet the attributes of a quality PRA.

2 (Page 15-5): The treatment of common cause failures in the IPE is noted as a concern based on the common cause database referenced in Chapter 14. This database has only recently been published (1995). Based on the state-of-the-art at the time the IPEs were performed, this data was not available (including passive components, such as heat exchangers and strainers) and a direct comparison to that database is not appropriate.

CHAPTER 16

1: Chapter 16 uses the NUREG-1150 analyses of early fatality and latent cancer fatality risk to make inferences about the IPEs. However, the additional insights gained from the containment performance evaluations and recent research in this area may lead to different conclusions than the 1150 analyses.

CHAPTER 17

1: The evaluation of the SBO rule would benefit from a review of the results by A/E and just not reactor type.

2 (Page 17-2): "Calloway" should be "Callaway."

4 (Page 17-11): Last paragraph. Second sentence, "Summer has a high LOSP initiating event frequency (0.073) and low EDG redundancy (i.e., two of three EDGs for safe plant shutdown following a LOSP)." Summer requires one of two EDGs for safe plant shutdown following a LOSP.

4 (Page 17-11): Last paragraph. Sentence "Do the units conservatively assume that the condensate..." looks like a question not a sentence. This is confusing.

5 (Page 17-[illegible]3): The discussion in the second paragraph and third paragraph. It implies [illegible] tripping the RCPs by the operators is important. However, this chapter deals with SBO. [illegible] SBO, the RCPs which are powered by non ESF buses are automatically tripped on a loss of offsite power and no operator action is necessary. This is a wrong discussion for this chapter.

VOLUME 1, GLOSSARY

1: The definition of "accident management" implies incorporation into the emergency response procedures. This definition should be embellished or use the industry definition.

2: The definition of "recovery action" is more focused than what the industry may call a "recovery action" in that alternatives other than restoring failed equipment are called recovery actions by the industry.

3: Overall, the glossary is a good first step to a common understanding between the NRC and the industry of PRA terminology.

VOLUME 2, ABBREVIATIONS

1 (Page xxxv): ARFS should probably be "Air Return **Fan** System".

2 (Page xxxvi): IPEP should be "Individual Plant **Evaluation** Partnership".

3 (Page xxxvii): NMLPCI should probably be LPCI.