ENTERGY

Entergy Operations, Inc.
P.O. Box 31995
Jackson MS 39286-1995
Tel 601 368 5760
Fax 601 368 5768

Jerrold G. Dewease
Vice President
Operations Support

March 7, 1997

Ms. Mary Drouin
Office of Nuclear Regulatory Research
Mail Stop T-10 E50
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Entergy Operations, Inc.
Comments Regarding Draft NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," October 1996

CNRO-97/00003

Dear Ms. Drouin:

The United States Nuclear Regulatory Commission (NRC) requested comments regarding a draft version of NUREG-1560. Entergy Operations, Inc. (EOI) appreciates the opportunity to provide comments on this draft document.

In general, EOI feels the overall intent and content of NUREG-1560 will form a valuable and long-lasting resource for use by both the industry and the NRC. However, the NRC needs to be aware of the potential for unintentional adverse consequences of pursuing some of the quality PSA attributes described in NUREG-1560. Some of these issues as described by NRC in the NUREG-1560 could result in a vast expansion of resources in order to support risk-informed regulatory decision-making. The resource commitment required to support these suggestions in the NUREG-1560 may be greatly out of proportion to the value added, which in many cases would be minimal. Further, the attributes described go well beyond what is necessary to support specific applications already being implemented at EOI and other utilities.

As the NRC considers the impact of risk-informed regulatory philosophy on specific techniques and methods, the NRC needs work with the industry to ensure that the new methods or requirements (1) are needed for the application intended and (2) add value which is commensurate with the associated increase in resources. Otherwise, new attributes and methods (such as those in NUREG-1560) could have the counter-productive effect of discouraging the use of PSA as a valuable tool to support safe plant operation.

In addition, some of the attributes raise potentially difficult technical questions. For example, the NUREG defines a "quality PRA" as including core damage frequency (CDF) due to internal and external events, and for at power, low power and shutdown conditions. It must be pointed out that combining these disparate values together or even comparing them is not as straightforward as it may seem. The maturity and state of the art of the types of analysis required for each of these different CDF estimates varies widely. There are different magnitudes and sources of uncertainties. The different analyses are based on different underlying assumptions and levels of conservatism. Simply combining them into a single value may be misleading given these facts. Even comparing the values may be a case of "apples and oranges".

EOI's detailed comments are included in the attachment to this letter. Please address any comments or questions regarding this matter to Bryan Ford at (601) 368-5792.

Sincerely,

Bryan Ford

for JGD/BSF/baa

cc: Mr. J. L. Blount
Mr. L. J. Callen
Mr. J. G. Dewease
Mr. J. N. Donohew
Mr. C. M. Dugger
Mr. J. J. Hagan
Mr. C. R. Hutchinson
Mr. G. Kalman
Mr. J. R. McGaha
Mr. C. P. Patel
Mr. D. L. Wigginton

The following comments, for the most part, are focused on NUREG-1560 Volume 1 Part 1. Many of these comments apply throughout the NUREG and the comments should be treated as applying wherever the discussed information is contained in addition to the specific instance(s) cited.

1. In the "Impact of the IPE Program on Reactor Safety" section of the Executive Summary, the report (page xi) states: "Although no common vulnerabilities were identified, the following vulnerabilities can be considered applicable to many BWRs ... Similarly, the following vulnerabilities can be considered applicable to many PWRs..." These statements should be changed to the following: "No common vulnerabilities were identified. The following vulnerabilities were identified at one or more BWRs ... Similarly, the following vulnerabilities were identified at one or more PWRs ..." The current statements are misleading by implying the vulnerabilities identified are generically applicable. As discussed in the report, no common vulnerabilities were identified by the licensees and the report itself does not provide the technical basis for identifying any common vulnerabilities. If a licensee did not identify an issue as a vulnerability at its plant with its plant-specific analysis, then that issue is **not** a vulnerability for that plant. The entire reason for the IPE program was the recognition that vulnerabilities are **not** generic but must be examined on a plant-specific basis. This same implication that the vulnerabilities and improvements identified are generically applicable can be found in several locations and should be removed unless the technical justification is provided (e.g., page 2-2 in the second paragraph and page 3-30 in the third paragraph, page 3-31 in the first sentence of the fifth paragraph). The most that can be said about the generic applicability of the plant-specific vulnerabilities and improvements identified is that the insights provided should be considered in the PSA studies at other sites.

2. In the first paragraph of the "Impact of the IPE Program on Reactor Safety" section of the Executive Summary (page xi), the report strongly implies that the reason that more plants did not identify vulnerabilities is that the individual plants defined the term "vulnerability" in such a way to prevent vulnerabilities from being identified. The paragraph also implies that the potential improvements identified in the IPEs are actually vulnerabilities. The report does not provide the technical basis for this implication and this implication should be removed or the technical basis provided.

3. In section 5.3 the last paragraph (page 5-17) states: "At least some of the variability in HEP values can arise as an artifact of the way in which HRA methods are applied. Nonetheless, the main point to be derived from examining the HEPs for specific actions across plants, is that, in most cases, it also appears that there are reasonable explanations for much of the variability in HEPs and in the results of the HRAs across the different IPEs." In contrast the report also states: "because many of the licensees failed to perform high-quality HRAs, it is possible that the licensees obtained HEP values that are not appropriate for their plants." The report does not provide the technical basis for determining that **many** of the licensees failed to perform high-quality HRAs in fact the report goes on to identify that "in most cases, it also appears that there are reasonable explanations for much of the variability in HEPs and in the results of the HRAs."

   In addition, in section 14.3.5, the characteristics of the HRA methodology included in a "quality PRA" are provided. As discussed above, in the NUREG the implication is clearly made that the IPEs did not include quality HRAs. However, the description of a quality HRA

in section 14.3.5 is almost identical to the process used in some IPEs. Combined with the discussion above this would tend to question the conclusion of the NUREG about the lack of quality in the HRAs performed for the IPEs.

The NRC should remove references to the lack of quality in the licensees' HRAs since the statements are not clearly supported by the facts in the report.

4. In the "Containment Performance (Containment Design) Perspectives" section of the Executive Summary, the report in the fourth paragraph compares the conditional failure probabilities of the BWR and PWR designs. This section says the BWR predicted failure probabilities are "quite high" and the PWR probabilities are "relatively low." The next paragraph then discusses radionuclides releases. The fourth paragraph comparison would be more balanced with the inclusion of the results of Figure 4.9, "Reported IPE conditional probabilities of significant early release by containment type". Figure 4.9 shows that although the conditional failure probabilities may in general be higher for BWRs than PWRs, the conditional probabilities of significant early release are not generally higher for BWRs than PWRs.

5. It does not seem appropriate to identify the HRA as an important shortcoming of the IPEs since the report identifies that in most cases there is a reasonable explanation for the plants HRAs. There are no "perfect HRAs" (and there probably never will be). In addition, that an HRA is perfect can not be proven since there will never be enough real accidents to develop a historical data base to validate the HEPs. Each plant's HRA must be judged on its own merit. A quality HRA is an HRA that has sufficient rigor for the application and is reasonably consistent throughout the study (i.e., the risk important systems would not change significantly even if an ideal HRA was performed). Please modify the report to identify that the HRAs are not an important shortcoming although they may be an area where additional work could be useful.

6. In the "IPEs with Respect to Risk-Informed Regulation" section of the Executive Summary, the report in the second paragraph states: "Many of the analyses relied heavily on either the use of the MAAP code or the use of a set of industry position papers, neither of which have a comprehensive treatment of severe accident phenomena." The report does not provide the basis for the determination that neither the MAAP code nor the industry position papers used by licensees provided a comprehensive treatment of severe accident phenomena. A more correct statement would be that the codes and industry position papers relied upon in the analyses have not, in all cases, undergone a rigorous review by the NRC and it is not assured that they provide a comprehensive treatment of severe accident phenomena. The fourth paragraph on page 6-17 also requires modification to address this issue.

7. In the "Overall Conclusions and Observations" section of the Executive Summary and on page 8-6 in section 8.2.4, the report states that the staff plans to pursue some type of follow-up activity for plants with relatively high core damage frequency (CDF) or conditional containment failure probability (CCFP). This does not seem to be justified considering the report identifies that when the IPE results are examined against the results of NUREG-1150 the quantitative health objectives are still being met for these plants.

The problem with the staff's approach is that it looks at CDF and CCFP as independent factors that can be scored separately. It assumes the existence of either a high CDF or high CCFP is evidence on its own of a potential concern. In reality, the two factors should be looked at together. They are each a part of the overall input to risk, which should be the figure of merit. For example, it is possible for a plant with a high CDF but low CCFP, a plant with a low CDF but high CCFP, and a plant with an average CDF and average CCFP to all have the same risk. The NRC should consider both the CDF and CCFP together when considering where to expend further resources.

8. In the "Core Damage Frequency (Reactor Design) Perspectives" section of the Executive Summary, the report in the second paragraph states: "the CDFs for many BWR plants are actually higher than the CDFs for many PWR plants." As shown by Figure E.1 the "many BWR plants" is at most 6 of the BWR plants. This statement would be more correct if "many" was replaced with "approximately 20 percent of the".

9. In the "Overall Conclusions and Observations" section of the Executive Summary, the report in the last bullet states the "IPE results indicate areas in PRA where standardization is needed." Standardization is not needed to close out the Generic Letter 88-20 review and is not required for other applications, although it is very helpful to reduce resource requirements. This bullet should be changed to "IPE results indicate areas in PRA where standardization would assist in the use of the IPEs for other applications ."

10. On Table 3.1 in the "Key Observations concerning ATWS," the second sentence states "BWR variability is mostly driven by modeling of human errors..." Also, the first sentence of the second paragraph of the ATWS discussion on page 3-23 states: "The ATWS results are affected more by modeling assumptions than by plant-specific design features." This conflicts with the finding of the report concerning HRAs. For example in Section 5.3 in the last paragraph the report states: "Nonetheless, the main point to be derived from examining the HEPs for specific actions across plants, is that, in most cases, it also appears that there are reasonable explanations for much of the variability in HEPs and in the results of the HRAs across the different IPEs." Also, in the "Human Action (Operational) Perspectives" section of the Executive Summary in the third paragraph, the report states: "in most cases there is little evidence that human reliability analysis (HRA) quantification method per se has a major impact on the results." Section 3.2.2 in the ATWS discussion on page 3-23 identifies three specific examples of the ATWS HRA differences. But it is not clear that these differences are an artifact of the modeling assumptions and not because of valid plant-specific design and procedural differences. Saying that the BWR variability is due to the modeling of the human errors instead of plant-specific features implies that the variability is artificial and not valid. This implication is not clearly supported by the report and should be removed or additional justification provided.

    A suggested solution to the Table 3.1 discussion is to replace "is mostly driven by modeling of human errors" with "is mostly driven by the HRA". On page 3-23 the sentence "The ATWS results are affected more by modeling assumptions than by plant-specific design features." should be deleted and in the next sentence "ATWS modeling features that" should be inserted after "key factors". Changes similar to those proposed for page 3-23 should be made to the discussion in the first paragraph on page 3-33.

11. Sections 3.2.1, 3.2.2 and 3.2.3 state "ISLOCAs, however, can be significant contributors to risk because the releases bypass containment." These sentences should be deleted because these statements are not supported for the BWRs (except for Big Rock Point) by the findings in sections 4.2.1, 4.2.2 and 4.2.3 which identifies that containment bypass is not important of the Mark I, II, or III designs. The phrase "are potentially important risk contributors (since the containment is bypassed)" should also be removed from the ISLOCA discussion in section 3.2.2 in the first paragraph on page 3-17 for the same reason.

12. In section 3.2.3 on page 3-29 the second sentence of the first paragraph says "... leaving only AC-independent systems (such as the steam-driven RCIC system and the HPCS system)..." The fact that the HPCS system requires AC power is discussed in the third sentence of the same paragraph. In the second sentence "associated with Division 1 or 2" should be inserted prior to the parenthetical expression and "and the HPCS system" should be deleted.

13. The assertion in Section 6.2 (as well as other sections) that a "quality" PRA is a state-of-the-art PRA is incorrect. This does not recognize the fact that the kind of "PRA analysis" performed and the needed level of detail and quality, may depend on the exact application. In many cases, a specific probabilistic analysis may be more appropriate than a full scope "PRA". The definition of a "quality PRA" will require input from both the regulator **and** the entire nuclear industry. The report should be revised to state that a "quality" PRA is a PRA of sufficient rigor to support the correct application of the insights provided by the PRA for the task at hand.

14. On page 6-12, the following sentence needs clarification: "Therefore, individuals who performed the PRA and other utility personnel are excluded from the peer review team." One interpretation is that other employees of the given utility are unsuitable "independent reviewers." Another interpretation would be that no employees of any utility should serve as reviewers for any PSA. This interpretation is supported by page 14-66 where the definition of the composition of the peer review team excludes any utility personnel including personnel from another utility that does not own the subject plant. This means that all true "peers" are excluded from being part of the "peer" review team. Please clarify the intention, and acknowledge that experienced PSA analysts working for utilities other than the one that performed the PRA would be excellent, independent reviewers. At the present time the BWROG IRBR committee is conducting an extensive peer review of a substantial number of IPE/PSAs. The BWROG PSA Peer Review Certification Pilot process has learned two valuable lessons that may be useful to the NRC in establishing their view of the state of the effort in assessing quality in PSAs. These two lessons are:

- Utility representatives from another utility who are PSA experts in their own right tend to be more critical of the host utility PSA than other independent reviewers (e.g., contractors).

- There are extensive lessons to be shared about PSA processes, methods, and data that is affected by the cross pollination of the PSA analysts from another utility reviewing the techniques of the host utility.

In summary, there are substantial benefits to be gained by encouraging peer review team to include PSA experts from sister utilities both to enhance the review of the subject plant and to further the general state of the technology through the sharing of superior approaches. Please modify the NUREG to acknowledge that experienced PSA analysts working for utilities other than the one that performed the PRA would be excellent, independent reviewers.

15. The discussion of the perceived shortcoming in the IPEs discussed in the last paragraph of page 6-14 and the first paragraph of page 6-15 needs to be qualified with the statements that the problems identified may not be actual problems with the model. The shortcoming identified may have been correctly addressed by the licensee but the documentation was not apparent in the information supplied with the IPE submittals.

16. On page 1-3, the draft document states that the NUREG was developed by relying solely on the information provided in the IPE submittals and without consideration of the accuracy of the submittals. However, the document repeatedly makes value judgments and criticisms of various aspects of both specific and general IPE submittals. This is particularly true of the discussions regarding human reliability analysis. If value judgments are going to be made they should be based on the best available information. The value of the NUREG comparisons and the accuracy of the document as a whole would be greatly enhanced if the NUREG compared the IPEs including the licensees' responses to the NRC's Requests for Additional Information and any other information that was provided to support the NRC's issuance of the Generic Letter 88-20 Safety Evaluations.

17. On page 6-3, it is stated that HRA is a special area requiring unique skills needs to be modified. Actually, all of the HRA methodologies seem to be intended to be applied by competent technical personnel. It may take special skills to develop the methodology, but it doesn't to correctly apply the methodology.

18. On page 6-2, a full scope PRA is defined as one including all events, both internal and external, under all plant conditions, including full power, low power, and shutdown. Does this mean the NRC is now intending to require all PRAs used in the regulatory arena to include shutdown risk assessments, full scope seismic risk assessment, and full scope internal fire risk assessment? This is a massive expansion in scope if it is what is intended, and like the assertion that a "quality" PRA is a state-of-the-art PRA, this does not recognize the fact that what kind of "PRA analysis" is done and what are its levels of detail or quality depends on the exact application. In many cases, a specific probabilistic analysis may be more appropriate than a full scope "PRA". The report should be revised to state that a "quality" PRA is a PRA of sufficient rigor to support the correct application of the insights provided by the PRA for the task at hand and that a full scope PRA is not always required.

19. Please clarify what is meant by Table 6.1 on page 6-5 where it is stated that plant-specific data should be quantified with a Bayesian update. Does this refer to updating a generic data distribution with the plant-specific data using Bayesian analysis techniques or does it refer to generating a distribution using only the plant-specific data using Bayesian analysis? Bayesian updating should refer to updating generic data with plant specific data.

20. On page 6-6 in Table 6.1 and in section 14.3.1, it is stated that a quality PRA would use a sufficiently low truncation limit so that 95 per cent of the CDF is captured. There is no way to prove or show that 95 per cent of the CDF has been captured. None of the current methodologies allows the determination of the total CDF, therefore, the CDF that is determined cannot be compared to the total CDF and a percentage developed. The general rule of thumb has always been to truncate at three or four orders of magnitude below the final CDF. It is assumed this will result in capturing about 95% of the CDF, but that cannot be proven. Please clarify this statement.

21. On page 6-8, it is stated that the minimum analysis attributes necessary to reflect the full range of views of the technical community can be defined. Since the full range of views of the technical community includes diametrically opposed views on some issues, it is very doubtful that this can be done for all issues. This statement should be clarified or deleted.

22. On page 6-8, it is stated the only way to address the uncertainty in containment performance is to assign and propagate statistical distributions. This ignores the fact that depending on the issue, sensitivity analysis may prove to be a better way of dealing with uncertain aspects. Modify this discussion to identify that the uncertainty in containment performance can be addressed in more ways then just assigning and propagating statistical distributions.

23. On page 8-3 in section 8.2.1, the NRC states that it plans to conduct follow-up activities to monitor implementation of the potential plant improvements identified by the IPE. The improvements were identified as "potential improvements". The NRC seems to be taking them as having been commitments. In most cases these improvements were identified only as areas for further review. These improvements should not be treated as commitments unless the utility clearly identified them as commitments.

24. On page 8-5 in section 8.2.3, the statement is made that although all licensees performed a PRA for the IPE, only a "small fraction" actually submitted the PRA. It should be pointed out that the NRC instructed that the PRA not be submitted as the IPE. NUREG-1335 identified the information the NRC wanted to see in the IPE. Most licensees adhered to this document's information request. It should also be pointed out that each licensee keeps all the supporting information, effectively the PRA, and this is available for NRC review. Keep in mind this NUREG is related to review of the IPE results, not PRA results.

25. On page 8-6 in section 8.2.3, the document states that the staff did not perform any verification or validation of the analysis or supporting calculations for the IPE/PRA. However, the staff draws conclusions regarding the validity of portions of the analysis, particularly HRA and data. These facts are inconsistent and the staff should perform any necessary verification and validation or the staff should clearly state that the conclusions presented are only opinion.

26. In section 8.4, the proposal is made to use the information gleaned from the IPEs and included in NUREG-1560 as the basis for focusing plant inspection activities. Although useful for broad inspection themes, this appears to be the use of second best information when looking at individual plants. Since each plant has a plant-specific IPE and PRA, these would seem to be the best source of information for use in planning and directing inspection activities. It should also be noted that many plants have updated or will update their PSA,

making the IPE and its conclusions in NUREG-1560 inaccurate. The NRC should endorse the use of plant-specific PRAs for optimizing and prioritizing plant-specific inspection activities.

27. In section 8.7, discussions regarding use of NUREG-1560 for a variety of issues is provided. However, most of the discussions are actually related to the use of the IPEs to address these issues. NUREG-1560 should not be the source of information for applications as discussed in the section 8.7. The IPEs/PRAs are the primary source and should be used. NUREG 1560 should reflect this.

28. In general in both chapters 5 and 13, the NUREG fails to discuss a very basic fact of how HRAs are performed and how this may impact the "variability" of values from plant to plant. Not all human error events in a PRA model are subject to the same level of analysis. The analysis begins with the use of "conservative" screening values. Usually, non-HRA dependent recovery events are applied next. The cutsets or sequences that are not truncated are then reviewed to identify the remaining human error events. These are subject to analysis, however, the level of detail is variable. If a conservative HEP can be applied that results in complete truncation of the cutsets or sequences containing the event or at least results in it becoming an insignificant contributor, then more detailed or refined analysis will not be performed. Likewise, if a combination of human error events appears in a cutset or sequence and the HEP for one of the events results in the truncation of the cutsets or sequences, then the other events may not require further detailed analysis and can be left with a "screening" value. These facts make it difficult if not impossible to take HEPs from one study and compare them to another. The level of detail required is dependent on factors that are totally plant specific. For example the arrangement of support systems is likely to result in a large variability in the importance of various sequences or cutsets from plant to plant. The NUREG should at least acknowledge these facts.

29. In section 14.3, the discussion of the attributes of a Level 1 PRA is totally dedicated to a linked fault tree model. The event sequence methodology is also a valid methodology. The same priority and level of discussion should be applied to the event sequence methodology. Note: this comment should be applied throughout the report.

30. On page 14-5 in section 14.3, the statement is made that two types of core damage probability are typically estimated. In general, core damage probability is not evaluated as part of a PRA. It is usually only used in a specific analysis to address a specific issue for risk management. In other words it is part of the application process of the PRA, but is generally not part of the PRA itself. Please clarify when core damage probability is expected to be used versus the use of core damage frequency.

31. On page 14-7 in section 14.3.1, the criteria used in a "quality PRA" for excluding initiating events includes truncating the event if its frequency is less than 1E-7/year. Any truncation value used in a "quality" PRA or even a "non-quality" PRA should bear at least some relationship to the final expected CDF and release frequency. Use of a pre-determined, generic truncation value may be inappropriate. This criteria should be modified or deleted.

32. In section 14.3.1, the discussion of the initiating event identification and grouping ignores one valuable technique. That is the development and use of a master safety logic diagram.

This oversight also seems to reflect a bias in the report toward linked fault tree analysis since MSLDs are generally only developed by the analyst involved in event sequence models. However, the MSLD is applicable to either method. Include a discussion of master safety logic diagram in this section.

33. On page 14-11, the discussion on modeling accident progressions clearly implies that human errors are modeled as event tree top events. For linked fault tree models, human actions may be modeled at the system fault tree level or at the sequence level (i. e., as an event tree top). They may also be applied after quantification. For event sequence models, the human errors are generally more likely to be modeled as event tree tops, although not necessarily so. The discussion should reflect this modeling.

34. On page 14-15, it is stated that for a "quality PRA" system models must include configurations that are not permitted by the technical specifications. This is never done. In addition, on page 14-3, the NUREG identifies as an assumption found in a "quality PRA", that the plant is operating within its technical specifications. Therefore, the NUREG is inconsistent on this issue. The reference to modeling all configurations including those in violation of technical specifications for a "quality PRA" should be deleted.

35. On page 14-16, the only acceptable source of common cause modeling and common cause failure data is identified as the AEOD report, which is actually in the form of an internal INEL report at the current time. This report has not been published as a NUREG. Therefore, it has not been subjected to a public comment and peer review. Endorsement of this report as the only acceptable methodology is pre-mature. The endorsement of the AEOD report should be removed.

36. On page 14-19 in section 14.3.4, it is stated that a quality PRA will use the number of generation hours for the plant from the NRC gray books. It is a better practice to go to the original source. In this case that would be the utility since the information in the gray books come from the utility. The statement should be deleted.

37. On page 14-20, the document defines the use of values lower than those in the AEOD report or excluding a common cause failure mode included in the AEOD report is inappropriate. On page 15-5 in section 15.2.3, it is stated that the AEOD report on common cause includes not only common cause failures among active components required to change state but also passive components including heat exchangers and strainers and clearly implying they should be modeled. Inclusion of common cause failures among passive components would be significantly beyond the current state of the art. What is the technical justification for these statements? The AEOD report is actually an internal INEL report at the current time. This report has not been published as a NUREG. Therefore, it has not been subjected to a public comment and peer review. Endorsement of this report as the only acceptable methodology is pre-mature. These statements concerning the inappropriateness of the values used in the IPEs should be removed unless additional technical justification is provided.

38. On page 14-20, the statement is made that a "quality PRA" uses Bayesian estimation to combine plant-specific data with generic data from AEOD reports. It is not clear what AEOD reports are meant. Also, what is the justification for using the AEOD reports? In addition,

the report defines a statistical methodology for both estimating the probability distributions for data and for performing Bayesian updating. The exact methodologies should not be included in NUREG-1560 but rather there should only be a general endorsement of the need to perform these tasks.

39. On page 14-21, the statement is made that for initiating event frequencies, only plant-specific data should be used and Bayesian combination with generic data should be avoided. Given the low scram rate, which continues to decrease, it would seem evident that the number of plant-specific scram events are likely to remain low, increasing the uncertainty generated by using only plant-specific data. What is the justification for not using Bayesian updating, properly of course, to try and reduce the uncertainty in the initiating event frequency determinations? This seems to be inconsistent with the treatment of component data. This comment also applies to the treatment of plant-specific test and maintenance outages.

40. On page 14-23, it is stated that only the data included in NUREG/CR-4550 and NUREG-1032 is acceptable for development of recovery probabilities and then only for the recovery of off-site power, PCS, diesel generators, and DC buses. No other data sources are considered acceptable. These statements are based on outdated information. EPRI has collected and published several excellent data sources, much more recent and robust than in NUREG/CR-4550 and NUREG-1032. These reports not only include data for recovery of offsite power and diesel generators, but also pump and valve recoveries such as those included in NSAC-161. The NUREG should acknowledge the fact that there are acceptable non-NRC sources of data and in fact many of them are of more recent and better quality.

41. On page 14-24, the statement is made that in a "quality PRA", all human events in the cutsets or sequences that do not truncate after initial quantification are quantified with the detailed HRA model. It should be noted that this could easily require the analysis of hundreds of events, comprising both single and multiple operator errors, depending on the how sequence or initiator specific the events may be. This is a significant burden on the analyst. Any shortcuts that do not invalidate the model, such as analyzing only one event in a group of multiple events since once its new HEP is determined the cutsets or sequences containing the group will truncate, should be encouraged and as a result this statement should be deleted.

42. On page 14-66 in section 14.7.2, the two bullets at the bottom of the page appear to be addressing reviews of proposed design changes, not reviews of PRAs. The wording should be changed.

43. On page 15-3 in section 15.2.2, it is stated that fuel cladding oxidation is the parameter used in Chapter 14 to determine the onset of core damage. However, on page 14-10, the onset of core damage for a "quality PRA" is defined as when no imminent recovery of coolant injection is anticipated and, therefore, a substantial amount of the radioactive material in the fuel gap is released. These two definitions should be reconciled or presented as alternatives.

44. On page 15-6 in section 15.2.4, it is stated that it is inappropriate to use one third of a failure as an upper bound for a no observed failure case for Bayesian updating. No technical

justification for this statement is provided. A technical justification for the use of the one third estimate is possible. A Bayesian analysis of the problem has shown that the use of one third provides an upper bound for a failure rate following an exponential distribution. The same analysis also showed that use of one assumed failure tended to be overly conservative and could result in invalid interpretation of plant specific data. This statement in the NUREG should be removed unless it can be technically justified.

45. On page 15-6 in section 15.2.4 as well as in chapter 14, it is stated that plant-specific data should be included for all initiating events and system components. This requirement seems to ignore the difficulty and resource requirements for generating that amount of plant-specific data. It should be perfectly adequate to use plant-specific data for the major components (e.g., ECCS pumps and diesel generators). A technical justification for why it is necessary to use plant-specific data for all components should be provided or these statements should be removed.

46. On page 3-31, the document states that only one BWR, Grand Gulf, credited CRD injection in the short term in their IPE submittal. This is incorrect. River Bend also credits CRD injection in the short term with two pumps if the vessel has not been depressurized. The statement needs to be revised to reflect this assumption.

47. On pages 3-32 and 9.25, the document states that RBS is considering increasing the size of the containment vent line. This is incorrect. This issue was addressed in the IPE submittal and the results indicated no favorable impact on CDF, and while it did decrease containment failures by 19%, it would increase large, monitored releases by 11.8%. Based on these facts, it was decided an increase in the size of the vent line would not be cost-beneficial since there was essentially no benefit. Therefore, RBS is no longer considering increasing the vent capacity. On page 9-25 in section 9.3.1.3, it is also stated that RBS is evaluating changes to the hydrogen igniter power source. Also, on page 9-41 in section 9.4.3.2, in the section on enhanced hydrogen igniter power supplies, it states that RBS states that portable DC generators to enhance DC power requirements can be provided. These statements are not true. River Bend evaluated this item in the IPE and determined that it did not provide cost-beneficial improvement. The NUREG section should be changed to reflect this. Note that in Table 9.12, this issue is correctly addressed. The table states that the RBS IPE, in responding to the CPIs, addressed the need for containment venting and hydrogen igniters and determined no modifications were deemed necessary. This is the true state of these issues regarding the River Bend IPE.

    With regards to enhanced power supplies for hydrogen igniters, the RBS IPE determined it would not provide cost-beneficial improvements in containment survivability to add a DC power supply. Instead, a portable AC diesel generator was added to provide power to either the division 1 or 2 DC bus (through the swing battery charger). This change results in a significant decrease in contribution of SBO to total CDF. It was determined that the reduction in SBO occurrence was more cost-beneficial than the minor impact on containment failure due to hydrogen igniter power supply upgrades.

48. On page 8-11 in Table 8.2, River Bend is credited with addressing Generic Safety Issues (GSI) GSI-23 and GSI-105. However, the RBS IPE also addressed the issues raised by Unresolved Safety Issue (USI) A-17 dealing with systems interaction. The IPE addressed

those portions of A-17 that it could, while the remainder of the close-out of A-17 awaited completion of the IPEEE. Therefore, USI-A-17 should be added to the list for River Bend in Table 8.2.

49. On page 9.25 in section 9.3.1.3, the statement is made that the implementation of the potential improvements identified in the RBS IPE for addressing SBO issues would result in a decrease in the SBO contribution to total CDF from 90% to 60%. Actually, the decrease would be from 86% to 54%. Please revise this statement.

50. On page 9-41 in section 9.4.3.2, the statement is made that the RBS IPE states that a cross-tie of the fire protection water to allow injection to the RPV was made subsequent to the IPE submittal. This is incorrect. A cross-tie from the FPW to the SSW system to allow injection to the RPV has always existed, been included in procedures, and was credited where reasonable for long term vessel injection in the IPE. The modifications to the system made on the basis of information gained in the performance of the IPE were merely to increase the ease of aligning the system and thereby decreasing the time required to perform the realignment. This should be corrected in the NUREG.

51. On page 10-9 in Table 10.2, the table should at least include a note referring to the fact that a major and unique decay heat removal system for RBS is the containment fan system. It should also note that RBS does not use containment spray.

52. On page 10-15 in section 10.2.1.5, the statement is made in the third paragraph that all other BWR 3s and 6s have multi-mode RHR systems that perform, among others, containment spray. This is not true for RBS. RBS uses a containment fan system instead of containment spray. Please revise this statement.

53. On page 11-41 in the discussion on RCIC failure modes, the statement is made that Perry models bypassing the steam tunnel temperature trip in the SBO evaluation. River Bend also credits the bypassing of the main steam tunnel high temperature trips since this action is an immediate action and, therefore, required to be memorized by the operators in the SBO procedure. This should be included.

54. On page 11-43, it is stated that the RBS ATWS model did not include two sequences that were shown to be significant at some plants. These sequences include failure of the recirculation pump trip and failure to inhibit ADS. This is not entirely true. These sequences were excluded from the RBS model quantification only after detailed, plant-specific analysis determined the impact of these two events was negligible. A detailed fault tree model of the reactor protection system for RBS was developed and quantified to determine the likelihood of failure of the mechanical and electrical portions of the RPS system as well as the failure of Alternate Rod Injection (ARI). These failure probabilities, when combined with the initiating event frequency and the estimated failure probability of RPT, resulted in frequencies below the 1E-08 per reactor year value used for the initial truncation. For failure to inhibit ADS, a detailed full THERP analysis determined the HEP for failure of the operators to inhibit ADS based on the RBS-specific PSFs was epsilon. Therefore, in both cases the sequences are indeed modeled, although only to the point at which the frequency of the sequence dropped below the truncation value. This is very different from not modeling the sequences. This point should be clarified in the write-up. In the same section, it is stated that the RBS model

assumes that the vessel level will settle at 2/3 core height with SLC failure but HPCS success. The wording seems to imply that this assumption may be invalid. It should be pointed out that this assumption is based on the information in the Grand Gulf NUREG/CR-4550 analysis (see NUREG/CR-4550, Volume 6, Revision 1, Part 1, page 4.4-109). If the assumption is valid for the BWR6 Grand Gulf then it should be valid for the BWR6 RBS.

55. The NUREG discusses uncertainty associated with the Byron Jackson N-9000 seals and infers that the IPEs are suspect in their RCP seal LOCA conclusions. Details concerning this technical issue have been provided to the NRC in various forums in the past. Details concerning the basis for the conclusions in the IPEs for RCP seal LOCA can be found in CRNO-91/00028, Letter from John R. McGaha (Entergy Operations) to Samuel S. Chilk (NRC), Solicitation of Public Comment on Generic Issue 23, 'Reactor Coolant Pump Seal Failure', and Draft Regulatory Guide; Issuance, Availability, Federal Register Volume Number 76 - April 19, 1991, dated September 27, 1991 and the NRC requests for additional information and subsequent replies associated with the IPEs. Please modify the NUREG to reflect the technical information provided in these documents and remove the inference that the IPEs are suspect in their RCP seal LOCA conclusions.

56. Section 9.3.2.1 refers to procedural changes for refilling the BWST during SGTRs at ANO-1. The guidance described in this section is actually referring to the Severe Accident Management Guidelines(SAMG) which are to be implemented. These SAMGs will utilize strategies for recovery efforts in the form of guidelines as opposed to procedures for implementation. Modify the discussion in the NUREG to reflect that the SAMG implementation instead of procedural changes provides the method for refilling the BWST during SGTRs at ANO-1.

57. Section 9.3.2.1 refers to CDF reductions for ANO-1 of 23% and 14% for AAC implementation and increased battery capacity. These values should be corrected to 36% and 16% respectively.

58. Table 11.16, "Other Features", and page 11-60 infer that an SBO diesel is credited in the ANO-1 IPE. The SBO diesel has been installed at ANO-1 since the IPE submittal and is not credited in any of the IPE results. Correct the table and discussions to reflect that ANO-1 did not take credit for an SBO diesel.

59. On Table 11.16 the ANO-1 SBO CDF is rounded to the next higher value of 2E-5 from approximately 1.6E-05. Correct the table to reflect the 1.6E-05 value.

60. On Table 11.18 correct the ANO-1 CDF to reflect a contribution from a large LOCA equal to 7.5E-07 and for small a LOCA equal to 1.49E-05.

61. Table 11.19 shows ANO-1 as having a relatively high spray initiation setpoint. However, the text on page 11-65 discusses that Crystal River has high setpoint spray initiation setpoint. Make these two discussions consistent.

62. Section 11.3.1.6 depicts ANO-1 as having the lowest CDF contribution for internal flooding. The value depicted in this section is 9E-07/ry. The ANO-1 internal flooding analysis was performed as a screening analysis. The CDF contributions for the flooding zones were

evaluated against the screening criteria of 1E-6/ry. For ANO-1, all zones fell below the screening criteria. Therefore, no specific CDF was calculated for the internal flooding analysis. Use of the value 9E-07/ry is inappropriate. Please modify the section to reflect ANO-1's use of screening criteria and remove the specific CDF value. Also, since no specific CDF value was calculated, the inclusion of an ANO-1 value for calculating an average CDF for B&W plants in incorrect and an ANO-1 value should not be included in the B&W average.

63. Section 11.3.1.6 expresses uncertainty in regard to ANO-1's treatment of breaks in the Service Water (SW) system in the internal flooding analysis and identifies that the SW is not an important contributor to internal flooding at ANO-1. In the ANO-1 analysis, for those zones containing SW piping, a guillotine break was assumed to occur flooding the zone for 20 minutes with a flow rate equivalent to two SW pumps in operation. The SW as a flood source is considered to be an important contributor to the internal flood analysis at ANO-1. Correct the NUREG to reflect the treatment of the SW system.

64. Section 10.2.2.5 states that ANO-2 has no PORVs. This is correct but misleading, since ANO-2 has an ECCS Vent Valve, which has a relief capacity greater than a PORV. Use of this valve allows RCS depressurization below the HPSI shutoff head. Please modify the NUREG to reflect the ANO-2 ECCS Vent Valve.

65. Table 11.22 indicates that HVAC failures were not considered in the ANO-2 IPE. This is not a correct statement: they were considered, but dismissed (via analysis) as contributors to core damage. Please correct the table.

66. Per Table 11.24, a "somewhat optimistic recovery" was applied to SBO CDF cutsets. In fact, the modeling of the recovery of offsite power is more appropriately described as "realistic", since it rigorously accounts for the timing of failures contributing to core damage relative the timing of the recovery of offsite power. The latter was based on a compilation of the best available industry data (NSAC and NUREG reports). Provide a basis for this statement or delete it.

Pacific Gas and Electric Company

245 Market Street, Room 937-N9B
San Francisco, CA 94105
Mailing Address
Mail Code N9B
P.O. Box 770000
San Francisco, CA 94177
415/973-4684 Fax 415/973-2313

Gregory M. Rueger
Senior Vice President and
General Manager
Nuclear Power Generation

March 10, 1997

PG&E Letter DCL-97-042

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to Request for Comments on Draft NUREG-1560

Dear Commissioners and Staff:

PG&E has reviewed the draft NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, Summary Report," in response to Mr. F. J. Miraglia's December 13, 1996 request for comments directed to my attention. PG&E's comments on the document are enclosed. These comments are limited in that they relate only to the accuracy of the draft NUREG as it pertains to the Diablo Canyon Individual Plant Examination (IPE). PG&E is coordinating general comments on the draft NUREG with the Westinghouse Owners Group and the Nuclear Energy Institute.

PG&E believes that draft NUREG-1560 reflects significant progress in the NRC Staff's efforts in the development of risk-informed regulation, and we look forward to publication of the final document.

If you have any questions regarding the enclosed comments, please contact Mr. Thomas Leserman at (415) 973-6530.

Sincerely,

Gregory M. Rueger

cc: Steven D. Bloom
Mary Drouin
James E. Dyer
Kenneth E. Perkins
Michael D. Tschiltz
Diablo Distribution

Enclosure

TLL/2225

~~9703180175~~ Dupe 2pp

PG&E COMMENTS ON DRAFT NUREG-1560, INDIVIDUAL PLANT EXAMINATION PROGRAM: PERSPECTIVES ON REACTOR SAFETY AND PLANT PERFORMANCE, SUMMARY REPORT

Table 11.39: Table 11.39 indicates that there are two diesel generators per unit at Diablo Canyon Power Plant, one shared diesel generator, and no cross-connects between units. After the Diablo Canyon Individual Plant Examination (IPE) Report was submitted to the NRC in 1992, a sixth diesel generator was installed at the plant; there are now three dedicated diesel generators per unit. Also, there is unit-to-unit electrical cross-connect capability at Diablo Canyon, although this capability is not credited in the probablistic risk assessment (PRA).

Table 11.39: Table 11.39 indicates that a "portable generator to provide continued AFW control" is an available feature at Diablo Canyon. As noted in PG&E Letter DCL-93-008, dated January 15, 1993, "Response to NRC Request for Additional Information on the Diablo Canyon Individual Plant Examination Report," question 34, "The CDF was determined to be relatively insensitive to this human action, as only a 0.2 percent increase in CDF was noted when this human action was guaranteed failed. Thus, this human action (ZHEHS2) was not proceduralized and the human action will be removed from the STADIC electric power recovery model as part of the next PRA update." We recommend deleting mention of the portable generator as an available capability at Diablo Canyon.

Table 11.42: Table 11.42 of draft NUREG-1560 states the Small LOCA initiator frequency for Diablo Canyon is 2E-3. The table also states that Diablo Canyon has Semi-Auto Recirculation switchover capability. Per Table 3.1.1-1 of the Diablo Canyon IPE, the Small LOCA initiator frequency should be 2E-2, not 2E-3. Also, a definition should be provided for manual, semi-automatic, and automatic recirculation switchover capability. Diablo Canyon has automatic residual heat removal pump trip at low refueling water storage tank level, but manual operator actions are required to complete recirculation switchover. Depending on the definition of manual and semi-automatic recirculation switchover, the Diablo Canyon switchover capability could be considered either manual or semi-automatic.

Table 12.13: Table 12.13 states that the Diablo Canyon ultimate pressure for containment was not provided (NP). Table 4.1-1, Item 14 of the Diablo Canyon IPE states the ultimate pressure for the Diablo Canyon containment is 168 psia, or 153 psig.

Table 17.3: The column "SBOR coping method" would be clearer if it were called "Action for SBOR compliance."