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ATTENTION: Dr. Medhat El-Zeftawy
US Nuclear Regulatory Commission
MS-P315
Washington, DC 20555

FROM: R. A. COSTNER RAC
DATE: October 18, 1993
SUBJ: COL CHANGES TO TIER 2 - OBSERVATIONS, COMMENTS, AND CONCERNS -- ADVANCED BOILING WATER REACTORS (GE) SUBCOMMITTEE -- ACRS/NRC
MEETING DATE(s): October 26-27, 1993
TOPIC: MEETING OF THE ADVANCED BOILING WATER REACTORS (GE) SUBCOMMITTEE -- ACRS/NRC

1. During the last months DOE and its M&O contractors have been placing a lot of emphasis on the process by which they control changes. An important part of this process is the review of each proposed change to determine if it may constitute an Unreviewed Safety Question (USQ). Although this has been a DOE requirement for a number of years, issue and implementation of DOE Order 5480.23 has brought increased attention and emphasis on the USQ issue and process.

2. The USQ evaluation process, like any old familiar process, looks different when applied to unfamiliar applications. Thus, for those of us familiar with USQ evaluations for commercial nuclear plants the recent emphasis on application of the process to other types of facilities resulted in some new insights.

3. For me a significant insight involved the <u>disconnect</u> between the USQ evaluation process and the PRA for a plant or facility.

4. I realized that for future commercial nuclear plants the use of PRA will be more and more important.

5. The GE Tier 1 document issued for review in the summer of 1992 (the last version received) contained Section 3.8, Reliability Assurance Program (RAP). The RAP was described as having the following two elements:

    (1) A Design Reliability Assurance Program (D-RAP) and
    (2) An Operational Reliability Assurance Program (O-RAP) and

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6. Concerning the O-RAP, Section 3.8 stated:

> The O-RAP is related to plant operating issues and will track equipment reliability to demonstrate that the plant is being operated and maintained consistent with Probabilistic Risk Assessment (PRA) assumptions such that overall risk is not unknowingly degraded during plant operation.

7. Concerning the D-RAP, Section 3.8 stated:

> The ABWR Design Reliability Assurance Program (D-RAP) is a program that will be performed during the detailed design and equipment selection phases of a project to assure that the important ABWR reliability assumptions of the PRA will be considered throughout the plant life. The PRA evaluates plant response to initiating events to assure that plant damage has a very low probability and that risk to the public is very low. Input to the PRA includes details of the plant design and assumptions about the reliability of plant risk-significant structures, systems and components (SSCs) throughout plant life.

8. For a commercial nuclear plant under conventional licensing having such a O-RAP program (based on such a D-RAP program) maintenance of the current validity of the plant PRA is very important. I would hope that the NRC would require commitments from the utility to evaluate each of the numerous plant changes made to such a plant for impact on the PRA. For changes which impact the PRA I would hope that the NRC would require commitments from the utility to continuously update the PRA in a timely manner.

9. The nature of such plant changes for a commercial nuclear plant under conventional licensing are well known. The majority of such plant changes are evaluated under 10 CFR 50.59, determined not to constitute a USQ, and implemented by the utility. The <u>more</u> significant plant changes are submitted to the NRC for review and approval.

10. The <u>most</u> significant safety-related concepts and issues for a commercial nuclear plant under conventional licensing are resolved and finalized in the FSAR and SER prior to the issuance of the OL. Therefore, the subsequent plant changes tend to be of lesser fundamental significance. As stated above, of those which occur, the more significant are required to be submitted to the NRC; those of lesser significance are evaluated by the utility under 10 CFR 50.59 and determined not to constitute a USQ.

11. Those submitted to the NRC and approved may or may not have the potential for significant impact on the PRA for the plant. At any event the NRC has an opportunity to suggest that the utility should evaluate the possibility of such impact.

12. Those evaluated by the utility under 10 CFR 50.59 and determined not to constitute a USQ and thus <u>not</u> submitted to the NRC may or may not result in evaluation by the utility for potential impact on the plants PRA.

13. For a commercial nuclear plant licensed under 10 CFR 52 the situation is somewhat different. The Tier 1 documents constitute the formal and legal regulatory definition of the safety bases for the particular standard plant design.

14. The SSAR (comparable to an FSAR) is considered to be a Tier 2 document. Changes to the Tier 2 document are to be evaluated by the COL applicant under 10 CFR 50.59 and, if determined not to constitute a USQ, implemented without NRC review.

15. Thus, for a commercial nuclear plant licensed under 10 CFR 52 the significance of a change evaluated under the rules of 10 CFR 50.59 may far exceed the significance of a change evaluated under the rules of 10 CFR 50.59 for a commercial nuclear plant under conventional licensing.

16. As a result the implications of a COL applicant's commitments to the 10 CFR 50.59 process with regard to the plant PRA are particularly important.

17. The following comments are intended to aid in consideration of what those commitments might include.

18. Most of the PRA / Fault-Tree work I have seen in the past are long on publishing results and short on documentation of the basis for the configuration being analyzed.

19. If a plant or a large facility performs a PRA, they generally end up with an expensive one-shot photo of their plant/facility.

20. As time passes modifications are preformed, and equipment ages, parts are replaced (other than a one-for-one basis), procedures are changed, etc. the PRA slowly becomes obsolete -- unless "maintenance" is performed to keep the mathematical model current. (The problem is similar to large complicated computer programs or models. This is a significant ongoing "maintenance" problem.)

21. Unless the PRA preparer prepares some extra aids or documentation (e.g., simplified sketches of mechanical and electrical configuration modeled) the operating staff or engineering support staff of the plant won't have a clear picture as to the exact plant configuration the PRA is based on.

22. As a result, when a modification is proposed, the operating staff or engineering support staff of the plant won't have any straightforward method to check whether that modification, etc. has a significant impact on the overall risk as indicated by the PRA.

23. I discussed the USQ evaluation aspect of this with associates experienced in preparation of such evaluations and have factored their comments into the items which follow.

24. If a plant/facility has a PRA they are relying on for overall risk assessment, there isn't any provision in the USQ evaluation process that would require (1) evaluating the impact of the modification on the overall plant/facility PRA model (2) updating that overall plant/facility PRA model.

25. As you know I personally have reservations concerning NSAC-125, the commercial nuclear power industry's guidelines on compliance with the 10 CFR 50.59 requirements. However, for the present discussion I will ignore those reservations and assume NSAC-125 presents the appropriate approach.

26. At present (without considering standard plant designs) there is no requirement to use PRA in the evaluation of a modification to determine if it constitutes a USQ.

27. If the modification is subsequently found not constitute a USQ and is implemented in the plant, there is no requirement associated with the USQ process to factor that modification into the PRA for the plant.

28. It's reasonable to suppose that a plant might do a before/after PRA on the particular modification to determine its impact. However, that's a long way from modifying the overall plant PRA.

29. As stated above, with the advent of commercial nuclear plant licensed under 10 CFR 52 the significance of a change evaluated under the rules of 10 CFR 50.59 may far exceed the significance of a change evaluated under the rules of 10 CFR 50.59 for a commercial nuclear plant under conventional licensing.

30. At the same time there is developing an increased reliance on the plant PRA for such programs as Reliability Assurance Programs (RAP) consisting of Design Reliability Assurance Programs (D-RAP) and Operational Reliability Assurance Programs (O-RAP).

31. Thus, maintenance of the current validity of the plant PRA in even more important than ever in view of the significance of changes permitted to be evaluated under the rules of 10 CFR 50.59 for a commercial nuclear plant licensed under 10 CFR 52.

32. Thus not only does the regulations need to require updating of the plant PRA but also there needs to be provision to document the PRA basis such that the COL applicant and/or plant/facility operating staff or engineering support staff can know that a particular modification <u>may have</u> a significant imp: .t on the plant PRA.

33. In addition there is no provision under present (or standard plant) rules associated with 10 CFR 50.59 for a commercial nuclear plant to evaluate individual modifications for synergistic (accumulated) impacts of the present and prior modification. (Each modification is evaluated against the current FSAR.)

34. <u>Summary of Concerns</u>

- The <u>disconnect</u> between the USQ evaluation process and the PRA for a plant or facility.
- The lack of requirements to evaluate the impact of modifications on the overall plant/facility PRA model.
- The lack of requirements to update the overall plant/facility PRA model if impacted by a modification.
- The lack of aids or documentation (e.g., simplified sketches of mechanical and electrical configuration modeled) to enable the operating staff or engineering support staff of the plant to know the exact plant configuration the PRA is based on.