July 30, 2001

Mr. Tony Pietrangelo
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SUBJECT: NEDC-32988, REV. 2, "TECHNICAL JUSTIFICATION TO SUPPORT RISK-INFORMED MODIFICATION TO SELECTED REQUIRED ACTION END STATES FOR BWR PLANTS" - REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MB1054)

Dear Mr. Pietrangelo,

By letter dated January 5, 2001, the Nuclear Energy Institute (NEI) submitted a technical report prepared for the Boiling Water Reactor Owners Group (BWROG) by GE Nuclear Energy (GENE). This report, NEDC-32988, Revision 2, "Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for BWR Plants," is relative to end states for technical specification action requirements. A request for additional information (RAI) is enclosed so that the NRC staff would have sufficient information to complete the review of the report.

If there are any questions, please call me at (301) 415-3016.

Sincerely,

/RA/

Robert M. Pulsifer, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 691

Enclosures:
1. Request for Additional Information
2. Proposed Additional Conditions

cc w/encls: See next page

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REQUEST FOR ADDITIONAL INFORMATION

BWR OWNERS GROUP TECHNICAL REPORT NEDC-32988, REV. 2

"TECHNICAL JUSTIFICATION TO SUPPORT RISK-INFORMED

MODIFICATION TO SELECTED REQUIRED ACTION END STATES FOR BWR PLANTS"

1. On page 3-1 it is stated that one of the primary considerations in allowing the vessel to stay pressurized in a hot shutdown mode (Mode 3) was the fact that Mode 3 "Achieves primary risk reduction by reducing source term/decay heat as the result of shutting down the reactor." What is meant by "primary" risk reduction? Does this statement imply a comparison between the risk of continued operation at power versus the risk of shutting down the plant? Please clarify this statement.

2. On page 3-1 it is stated that one of the primary considerations in allowing the vessel to stay pressurized in Mode 3 was that this “Avoids challenging the shutdown cooling system and associated potential for personnel errors.” Please identify important personnel errors associated with the alignment of the shutdown cooling system which are modeled in the probabilistic risk assessment (PRA) used to perform the risk assessments mentioned in the report.

3. At the beginning of page 4-2 it is stated:

   "The core damage frequencies (CDFs) associated with being in Modes 3 and 4 was evaluated. Then, for both modes, the increase in CDF associated with the specific equipment/system being unavailable was calculated. The relative CDF change provided an assessment of which mode is a safer state to be in when the equipment/system is unavailable."

   The results of these calculations were not included in the submitted report. This information is needed to finalize the review and prepare a safety evaluation. Please submit this information. Also, please clarify the last sentence in the above statement by defining the term "relative CDF change." It appears that, in general, both the base case CDFs (associated with the two Modes) as well as the CDF changes (when the equipment is assumed unavailable) are needed to determine which mode is the safer state to be in when equipment is unavailable.

4. On page 4-2, under General Assumptions, it is stated that "The plant is assumed to be shut down to handle a specific LCO item and, consequently, it is assumed that no additional systems are unavailable due to maintenance at that time." However, it is not clear what prevents some redundant trains of other systems to be unavailable for maintenance. Please explain.

5. On page 4-2, under General Assumptions, it is stated that "The PSA models [for] both Mode 3 and Mode 4 are for steady states." Does this imply that no transition risk between Modes 3 and 4 was considered? Was credit taken in the risk assessment for

the low pressure systems, as defense-in-depth when the high pressure systems are unavailable or fail, by depressurizing the reactor? Similarly, once core cooling fails in Mode 4, the reactor pressure increases and high pressure systems, such as high pressure coolant injection (HPCI) and reactor core isolation coolant (RCIC), can be used to cool the core. Obviously, the use of these "defense-in-depth" systems involve transition risks. Please explain how these "defense-in-depth" systems were credited in the risk assessment and what assumptions were made regarding failures (e.g., human errors) during the transition from Mode 3 to Mode 4 and vice versa.

6. On page 4-3 it is stated that the control rod drive (CRD) system "can be a significant contributor to core cooling in Mode 3." Please clarify this statement by referring to the success criteria and related assumptions of using the CRD system to cool the core in Mode 3.

7. On page 4-3 there is a short discussion about depressurizing the reactor in order to use low pressure systems while in Mode 3. It is stated that "the Emergency Operating Procedures (EOPs) require the operator to depressurize the reactor manually; however, if the operator does not depressurize in time, the Automatic Depressurization System (ADS) will be initiated automatically." Please explain how depressurization works in the various BWR designs (e.g., systems, success criteria, procedural requirements, actuation type) and how it is modeled in the risk assessment.

8. On page 4-4 it is stated: "Based on this qualitative comparison, it is judged that there are more systems that can provide this function in Mode 3 than in Mode 4." However, such a statement is not clearly supported by the two sets of systems listed on page 4-3. Please explain.

9. One of the steps in developing the Mode 3 and 4 PRA models (listed on page 4-5) involved the identification of initiating events (IEs). However, the selected IEs are listed on page 4-6 without adequate explanation on how they were derived. This information is needed to ensure that all relevant IEs have been considered. An acceptable approach is to start with an initial "generic" list (which includes all postulated types of IEs) and screen out those that either do not apply in Modes 3 and 4 or are unimportant based on bounding qualitative arguments. Please provide this information.

10. The report contains conflicting statements regarding the performance of sensitivity studies to address the impact of uncertainties associated with certain IE frequencies. On page 4-6 it is stated that sensitivity studies were conducted to ensure that the conclusions are not sensitive to assumed IE frequencies. However, on page 4-9 it is stated that such sensitivity studies were not performed because the "factor increase in both models is the same." In addition, the staff does not agree with the latter statement. In general, one would expect that a certain increase in an IE frequency (e.g., loss of offsite power) would impact the two models differently. This is due to the fact that some IEs are applicable to only one of the two modes. Also, in addition to causing a reactor trip, many IEs have an impact on accident mitigating systems which can be different for the two models. Please discuss potential uncertainties associated with IE frequencies and, if necessary, perform sensitivity studies to assess the impact of such uncertainties on the conclusions of the analysis.

11. On page 4-7 it is stated that "[n]ew event trees were developed based on the success criteria developed for Modes 3 and 4." Please submit these event trees for staff review, including related top event descriptions and a brief discussion of success criteria and important modeling assumptions.

12. On page 4-8 it is stated: "Because of the very limited time involved in this transition and the low probability of drain-down when the RHR system is engaged in the SDC mode, it is judged that the transition risk from Mode 3 to Mode 4 is insignificant relative to overall Mode 3 and Mode 4 risks." The staff does not believe that, in general, a plant transition does not involve significant risk if it takes place in a very short time. On the contrary, a transition that involves alignments requiring human actions can involve significant risk independently of how short the transition is. Please clarify the above statement and explain the reasons for the low probability of drain-down when the RHR system is engaged in the SDC mode.

13. It is stated (Section 4.4.3.3; page 4-8) that "...the conclusions reached in this report, based on the internal event PSA, remain unchanged even when external events are factored in." Qualitative arguments are used to support the above statement for seismic as well as for internal fire and flood events. With regard to internal fire and flood events, the main argument is that such events "....are generally more likely to occur during Mode 4 because of increased maintenance activities and presence of transient combustible material in Mode 4." The staff does not agree that increased maintenance activity takes place every time the plant is being shut down by the actions specified in the technical specifications (TSs). On the contrary, what is more likely to happen is to keep the plant in the "end-state" allowed by the TSs long enough to fix the problem and then return to power operation again. A comparison between Mode 3 and Mode 4 with respect to the impact of a fire or flood event on the systems used in these two modes would be an acceptable qualitative argument. Please explain.

14. Section 4.4.3.4 discusses several sensitivity studies performed to assess the robustness of the results to uncertainties in data and modeling assumptions used in the PRA. The staff requests more information regarding the approach used to identify important uncertainties in data and modeling assumptions as well as information regarding the numerical findings of the sensitivity studies. Please provide the following information:

- A brief discussion of the approach used to identify potentially important uncertainties in data and modeling assumptions used in the PRA.

- The initial list of potentially important areas of uncertainty in data and modeling assumptions that were identified.

- Discuss the qualitative arguments used to justify screening out any uncertainties from further evaluation through sensitivity studies.

- Results of sensitivity studies, preferably in tabular form, showing assumed changes in parameters, assumptions or models and appropriate interpretation of results in terms of the assumed changes.

15. Section 4.4.3.4 discusses several sensitivity studies performed to assess the robustness of the results to design and operational differences among BWR plants. The staff requests more information regarding the approach used to identify important design and operational differences (i.e., differences which have the potential to change the results and conclusions of the analysis) as well as information regarding the numerical findings of the associated sensitivity studies. Please provide the following information:

    - A list of potentially significant differences in design and operational features, preferably in tabular form, of the various BWR plants with respect to the plant model used in the analysis.

    - Discuss the qualitative arguments used to justify screening out any differences in design and operational features from further evaluation through sensitivity studies.

    - Results of sensitivity studies, preferably in tabular form, showing assumed changes in design or operational features and appropriate interpretation of results in terms of the assumed changes.

16. In the Abstract on page iii/iv, it concludes, "the proposed end state improvements provide more systems and operational flexibility while avoiding risk sensitive cold shutdown required actions and alignments." Similarly during hot shutdown, plant staff must take actions and perform equipment alignments. The "defense in depth" considerations submitted to support the change in end state from cold shutdown to hot shutdown is not sufficient. The plant can be in a hot shutdown for an indefinite period of time. Additional single failure or operator error can be postulated during this indefinite period. Evaluate potential Chapter 15 transients and accidents which can be postulated during hot shutdown, postulate possible single failure or operator error for possible scenarios during hot shutdown and then compare the consequences for hot shutdown and cold shutdown conditions. Submit a table showing the results.

17. According to the Introduction/Background section of the topical report, the assessment considered all TS end states resulting in a cold shutdown (Mode 4). The analysis then evaluated an alternate end state of remaining in hot shutdown (Mode 3) as a preferred alternative to Mode 4. Other potential preferred alternatives, such as low power operation, were outside the scope of this study. Based on staff review of the topical report, changes to TS 3.0.3 end states and reactor steam dome pressure end states, i.e., maintaining reactor pressure above 150 psig, were also proposed by the report. These changes were not discussed in the Introduction/Background section. Additionally, these changes are not consistent with the proposed changes by the CEOG topical report on end states. Further discussion and justification of proposed changes to TS 3.0.3 end states and reactor steam dome pressure end states are needed.

18. The BWROG topical report proposed adding additional conditions to several TS. However, the topical report did not discuss what the additional conditions would be. Descriptions of the proposed additional conditions should be provided in the topical report. This will also aid the development of the TSTF which would support the topical report. Examples of the proposed additional conditions was provided during the

June 27, 2001 meeting with the TSTF Task Force (enclosure 2).

19. Staff guidance for plant maintenance is given in Regulatory Guide (RG) 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." The RG endorses NUMARC 93-01, especially Section 11. Confirm that the topical report complies with Section 11 of NUMARC 93-01. Identify, if any, inconsistencies with Section 11 of NUMARC 93-01.

20. During hot standby, power conversion system (steam path through MSIVs to balance of plant [BOP] and condensate) is used as the normal heat sink. Discuss the alternate heat removal methods in case the normal heat sink is lost during transients and accidents while at hot standby.

21. On page 4-3 of NEDC-32988, critical safety functions are described. However, there is no mention of electrical power supply and the support systems required for other critical safety functions. Please provide appropriate information.

22. Section 4.4.2.3, Core Decay Heat Removal. LPCS is described as low pressure cooling system. Typically LPCS abbreviation is used for low pressure core spray system.

    Why isn't high pressure core spray included (HPCS)? HPCS is designed to function during reactor low pressure and high pressure?

23. Section 4.5, "Individual Technical Specification Assessments." Submit marked-up copies of all the proposed TS changes including the TS Bases.

24. **BWR/4**, Section 4.5.1.2, LCO 3.4.3, Safety/Relief Valves - Specify the actions for C1 and C2, also the completion times for both.

25. **BWR/4**, Section 4.5.1.3, LCO 3.5.1, ECCS System (Operating) - Add G to Conditions C,D,E,F in the new proposed Condition H2.

26. **BWR/4**, Section 4.5.1.6, Low-Low Set (LLS) Valves - Specify the actions for the proposed C1 and C2 and the duration times.

27. **BWR/6**, Section 4.5.2.2, LCO 3.4.4, Safety/Relief Valves (SRVs) - Specify the actions for C1 and C2 and the completion times for both.

28. **BWR/6**, Section 4.5.2.5, LCO 3.6.1.6, Low-Low Set (LLS) Valves - Specify the actions for the proposed C1 and C2 and the duration times.

BWR/4 PROPOSED ADDITIONAL CONDITIONS

Scope: "The assessment considered all Tech Spec end states resulting in a cold shutdown (Mode 4). The analysis then evaluated an alternate end state of remaining in hot shutdown (Mode 3) as a preferred alternative to Mode 4. Other potential preferred alternatives, such as low power operation, were outside the scope of this study."

Based on staff review - TS 3.0.3 end states were considered in this review although it was not specifically discussed in Introduction/Background. CEOG did not propose changes to TS 3.0.3 end states.

3.4.3 Safety/Relief Valves

Proposed Modification: Delete Required Action B.2 and add Condition C with Required Actions C.1 and C.2 to address three or more required SRVs inoperable.

Comment: No discussion of proposed Condition C. Proposed TSTF change?

3.5.1 ECCS System (Operating)

Proposed Modification: Delete Required Action B.2. Renumber Condition H (and Required Action H.1) to Condition I (and Required Action I.1). Renumber Condition G (and Required Actions G.1 and G.2) to Condition H (and Required Actions H.1 and H.2) and remove the "OR" condition. Add a new Condition G that is similar to the existing Condition G but with the first condition ("Two or more ADS valves inoperable") deleted and Required Action G.2 deleted.

Comment: Deleting requirement G.2 does not appear to be risk-informed. Proposed TSTF change?

3.5.3 RCIC System

Proposed Modification: Delete Required Action B.2.

Comment: Doesn’t meet the definition of end state as discussed in the background section of the topical report.

3.6.1.6 Low-Low Set (LLS) Valves

Proposed Modification: Delete Required Action B.2 and add Condition C with Required Actions C.1 and C.2 to address two or more LLS valves inoperable.

Comment: No discussion of proposed Condition C. Proposed TSTF change?

3.6.1.7 Reactor Building-to-Suppression Chamber Vacuum Breakers

Proposed Modification: Modify Condition E to relate only to Condition C and delete Required Action E.2. Add Condition F with Required Actions F.1 and F.2 to address the required actions related to Conditions A, B, and D.

Comment: No discussion of proposed Condition F. Proposed TSTF change?

3.6.1.8 Suppression Chamber-to-Drywell Vacuum Breakers

Proposed Modification: Delete Required Action C.2 and add Condition D with Required Actions D.1 and D.2 to address Condition B.

Comment: No discussion of proposed Condition D. Proposed TSTF change?

3.6.2.3 Residual Heat Removal Suppression Pool Cooling

Proposed Modification: Delete Required Action B.2 and add Condition C with Required Actions C.1 and C.2 to address two RHR suppression pool cooling subsystems inoperable.

Comment: No discussion of proposed Condition C. Proposed TSTF change?

3.7.2 Plant Service Water System and Ultimate Heat Sink

Proposed Modification: Delete Required Action E.2 and add Condition F with Required Actions F.1 and F.2 to address the remaining portion of Condition E.

Comment: No discussion of proposed Condition F. Proposed TSTF change? STS Rev. 2 different than STS Rev. 1 (so Conditions do not match).

3.7.4 Main Control Room Environmental Control System

Proposed Modification: Delete Required Action B.2. Change Required Action D.1 to "Be in Mode 3" with a Completion Time of "12 hours."

Comment: STS Rev. 2 different than STS Rev. 1 (so Conditions do not match).

3.8.4 DC Sources

Proposed Modification: Delete Required Action B.2.

Comment: STS Rev. 2 different than STS Rev. 1 (so Conditions do not match).

Similar comments on proposed changes to BWR/6.