July 31, 1992

Docket Nos. 50-369
and 50-370

Mr. T. C. McMeekin
Vice President, McGuire Site
Duke Power Company
12700 Hagers Ferry Road
Huntersville, North Carolina 28078-8985

Dear Mr. McMeekin:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING MCGUIRE INDIVIDUAL PLANT EXAMINATION (TAC NOS. M74430/M74431)

The NRC staff is reviewing Duke Power Company's Individual Plant Examination (IPE) submitted for the severe accidents assessment for McGuire Nuclear Station, Units 1 and 2. We find that we need additional information in order to complete our review of the internal events analysis in the IPE, the containment performance improvement program, and Generic Issue 105, "Interfacing LOCAs at LWRs." Please respond to the questions identified in the enclosures within the previously agreed upon 60-day, 90-day, and 120-day response schedule (from receipt of this request). If you have any questions concerning this request, please contact me at (301) 504-1479.

This request affects fewer than 10 respondents and is, therefore, not subject to Office of Management & Budget review under P.L. 96-511.

Sincerely,
/S/
Timothy A. Reed, Project Manager
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Division of Reactor Projects-I/II
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As stated

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QUESTIONS ON MCGUIRE INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL

1. In Section 3.3. [Flood Analysis] your IPE submittal explicitly states that "the only failure mode addressed in this analysis is submergence." NUREG-1335 reporting guidelines also identifies "water intrusion" as another failure mode. Based on previous plant operating experience, NUREG-1174 also identifies other failure modes including water spraying, dripping, or splashing on sensitive equipment. Please discuss other failure modes your IPE considered, including the potential for inter-zone flood propagation associated with internal flooding.

2. Your IPE states that there are two safety valves on the pressurizer whereas your FSAR states there are three. Please clarify.

3. Your IPE is not explicit on the ability of McGuire Units to cope with station blackout. Please discuss the IPE's treatment of systems important for coping with station blackout, including inter-unit ties for situations where one unit is blackout, and associated system limitations with time. Describe any credit taken for related operator actions.

4. Instrument tube failures were not explicitly addressed in the McGuire IPE, but are of potential concern because of their location. Please describe the process used to consider and screen out instrument tube failure(s) as an initiating event.

5. Your transient event tree model does not appear to address overcooling events. Please describe the IPE's treatment of overcooling events and ability to cope with adverse overcooling events. Consider in your description differences between steam line break initiators and feedwater line break initiators.

6. Provide a specific reference documenting the success criteria for feed and bleed.

7. The success criteria for LOCA's appear to address the injection phase only; however, the event tree models include recirculation. Please verify that the success criteria table (Table 2.3-1) included the recirculation phase and impact of support system failures.

8. Your submittal states that "A series of MAAP computer runs were performed to determine success criteria for various scenarios resulting from a small break LOCA. As long as SSHR is available, makeup flow from one of four SI or CVCS pumps prevents core damage." Given a two inch Medium LOCA and only one SI pump (no steam generator heat removal), it appears that the accident cannot be successfully terminated because primary pressure rises and prevents ECCS injection. Please discuss further the use of MAAP to develop success criteria in the IPE process, explicitly for 2" LOCAs case where only one pump is available for successful mitigation.

9. Your IPE states that the success of the SI (safety injection) System does not require switchover to hot leg recirculation. Switchover, however, is believed to prevent boron precipitation in the bottom of the core which could otherwise block core cooling water flow. Please clarify the observation that it is not necessary to model switchover to hot leg recirculation.

10. The SGTR modeling and the associated event tree appear incomplete in that they do not discuss (or model) the potential need for primary depressurization in situations where the steam generator with the failed tube cannot be isolated, e.g., stuck open SG safety relief valve. Please discuss the IPE treatment of this potential scenario.

11. Air operated valves 1RN89A and 1RN190B control the flow rate of nuclear service water through the CCW heat exchangers. The modeling of valve 1RN89A appears to be inconsistent with the requirement that this valve be fully open under accident conditions, unless, in normal operation, valve 1RN89A is always throttled fully open. Please verify that valve 1RN89A does not have to fully open under accident conditions, or discuss the rationale for treatment in IPE.

12. As per NUREG-1335 submittal guidance, provide the dependency matrices for all support systems and front-line systems considered (for each unit if they are different). Also include dependencies caused by systems that are shared among multi-unit plants

13. Identify your source of common mode failure data, and discuss the extent to which common cause failures were considered in the system fault trees. We noted that you have excluded common cause failure of check valves in the interfacing LOCA analysis. Please include this exclusion in the above discussion.

14. If recirculation is lost once the containment sump saturates, then core cooling during the recirculation phase of a LOCA following melting of all ice in the ice condensers requires containment spray operation in addition to heat removal through the RHR service water heat exchangers. Please discuss this apparent dependency among core cooling systems and containment heat removal systems (or lack thereof).

15. The IPE does not describe any dual-unit core damage scenarios and leaves the impression that dual-unit core damage scenarios are due solely to combinations of independent core damage events. Please describe how the McGuire Unit 1 and 2 IPE treated multi-unit scenarios, i.e., initiating events that could affect both units simultaneously.

16. Clarify the characterization of your component failure data and overall core damage frequency, e.g., mean, median, or point estimate.

17. Discuss the approach used in selecting your IPE t [illegible] ition measure used in capturing all 1E-7/yr sequences.

18. Describe the recovery action for T14 (loss of vital bus) initiated sequences and/or other plant features which had a positive impact with regard to the significance of the sequence.

19. Table A.19-1 states that failure of the HVAC in switchgear room had not been specifically analyzed, but eliminated by quantitative analysis, "see Reference 5." Please provide Reference 5 as it had not been included in the References Section (A.19.18.1).

20. With regard to your quantification of component failures, describe the process used to treat plant specific data as per NUREG-1335, Section 2.1.5.5.

21. Since the staff was not able to find an adequate documentation of SHARP tasks actually accomplished, explain how the following steps were accomplished as described in chapter 5 of the PRA:

    a) "STEP 1: Select and train team."

    b) "STEP 2: Familiarize with the plant." Discuss this process, including the number of plant visits performed, the objective of the visits, the team members participated, and tasks performed during the visits including the collected list of human-machine characteristics, the general human factors determined to guide the qualitative analysis.

    c) "STEP 3: Build an initial model." Provide a list of a) pre-initiator events (or latent, type 1 events) that were initially collected and evaluated based on McGuire operating history; b) human-induced initiators (or type 2 events) based on McGuire operating history that were initially considered; and c) post-initiating events, i.e. human activities in response to an off-normal condition (type 3 through 5 events) based on McGuire operational and training (classroom as well as simulator) experience that were initially considered.

       Provide a list of credible (latent or dynamic) events that were identified at the beginning of systems analysis as credible latent events and were numerically screened and explain on what basis the screening values (of .01 and .1) were chosen. Verify that no sequences were eliminated due to the chosen screening human error probabilities.

       Identify common cause failures due to human errors and discuss briefly how they were treated.

    d) "STEP 4: Characterize human interaction events." Provide a sample of descriptive or graphical representations of the breakdown of initially identified events into failure modes, mechanisms, causes, and effects (like FMEA).

    e) "STEP 5: Screen human interaction." Provide a list of human events that were initially considered, the assigned screening values (including the basis for), and the final list of human events and corresponding values.

f) "STEP 6: Quantify human interaction." Provide the data used (plant specific human errors, procedures, training experience, and simulator data) and explain how the quantification was performed by using steps 1 through 5 above.

Specifically, the staff was not able to identify any explicit specification and evaluation of the significant performance shaping factors (PSFs) affecting the actions (or classes of actions) analyzed in the IPE. Please provide a description of the PSFs considered, the rationale or a systematic process by which they were evaluated and selected, and any plant specific evaluation of those factors, and conclusions from this assessment.

g) "STEP 7: Identify recovery events." Provide a list of the recovery actions considered and the related bases (such as procedures, training, simulator ability to model these actions, etc.). Please provide cross references with the corresponding event trees/fault trees to enhance the review process.

h) "STEP 8: Update plant model." Identify the final plant model "that includes all human events important to objectives of the analysis."

i) "STEP 9: Review of results." It is stated that "the output of this step is a human reliability analysis that reasonably reflects the contribution to plant risk of the human interactions," (PRA, page 5.3-6). Please provide your findings.

j) "STEP 10: Document Analysis"

It is stated (pg. 5.3-6 of PRA): "Each step of the HRA process should be documented as well as the following:

The specific operator actions explicitly modeled.
The types of operator actions modeled implicitly.
The nature of the operating history review.
The method of screening and quantitative values, if used.
List of important actions carried on for more detailed analysis.
The nature of plant specific interviews.
Operation history reviews applied.
The type of control room reviews.
Reasons for selection of methodologies used.
The nature of operator actions performed."

Describe concisely the documentation discussed above.

22. Regarding the HRA assumptions (page 5.5-1):

   a) Define "fair credit" for "well trained people," i.e., what was the basis for the numerical estimates were used. Compare your fair credit approach vs. the SHARP recommended approach for well trained people.

   b) Explain "inadvertent actions are generally recoverable..." and "inadvertent actions during an incident are not modeled because of the human redundancy that would exist then." However, the objective of the HRA is to identify potential for human error and assess its contribution to risk. In fact, research on human error shows that analysis of "recoverable" errors provides the means for avoiding non-recoverable errors (e.g. see chapter 10, " General view of accident causation in complex systems, "of <u>Human Error</u>, by J. Reason, Cambridge University Press, 1990). Because of your assumption, a major class of human errors is eliminated from the analysis. Provide a basis for such an approach (based on plant operational history). Compare the IPE approach for inadvertent actions to SHARP recommended approach for inadvertent actions.

   c) Explain assumption #5.

23. Although it is stated that SHARP is used for the HRA, a big portion of the SHARP analysis is omitted in evaluating latent errors and instead a "generic" THERP method is adopted "since extra expense in detail analysis does not produce a commensurate gain in precision" (page 5.6-1). Thus, instead of THERP, an in-house derived, factored model (eq. 5.6-1) is used. The IPE does not provide a basis this model nor does it provide an evidence that the model was peer reviewed for its suitability in quantifying human error by HRA experts.

   Provide a basis for equation 5.6-1 as appropriate latent human error quantification and explain what was learned, by adopting this method, for improving human performance in the plant.

24. Provide a list of dynamic failure events that were identified as credible events at the beginning of the systems analysis and were numerically screened. Provide, as SHARP requires, an example of the more detail analysis performed for the events that were further analyzed including their breakdown by defining the subtasks to be performed, and the influence factors associated with the event such as the primary human behavior content, the subtask to be performed (e.g. reading), the stress level, the environment, the time allowed for the task, and the time allowed in the simulator. Provide the "representations" that were employed to structure the events under consideration, in terms of the OAT technique or any other representation technique that was used for this part of the analysis. Identify (as appropriate) the set of modified system event trees with human interactions derived for final quantification. Provide the data base used for the quantification which is described in section 5.6.3.

For example, the probabilities assigned to the group of events representing a failure to recover off-site power, before core uncovery following a loss of off-site power (page 5.7-2) has a large impact on the IPE core damage frequency calculations due to LOOP initiating event. The values assigned to these events vary from E-1 to E-3. The report states that these values were based on "historical data which was input to a power recovery model." Explain the model and the data used for the quantification including the failure data, the procedures available for recovery actions, and the time considerations taken into account in the quantification. Address these issues in greater depth for the sequences involving station blackout conditions.

25. The staff was not able to identify sensitivity analysis as part of the HRA process, although included in the SHARP process. Performing a sensitivity analysis is especially important to the Mcguire IPE because virtually every sequence identified as a significant contributor to risk has critical human actions and because human recovery actions play a significant role in reducing the estimated risk. Please explain.

26. The staff was not able to identify a concise definition of "vulnerabilities," and a clear discussion of how they were identified and addressed. Please explain.

27. Several front-end interface issues could not be resolved using the information available:

    a) Justification for the assertion that operators can locate and isolate the sources of flooding within ten minutes.

    b) Credit taken for the operator tripping the reactor coolant pumps during a small LOCA.

    c) Consideration related to operator actions for preventing boron precipitation given that long-term switchover to hot leg injection during a LOCA.

    Provide information needed to address these issues.

28. Several back-end interface issues could not be resolved using the information available:

    a) The basis for the engineering judgement for the operator action ZWLM221DHE. Also, is this the only operator action considered for containment isolation following an isolation failure?

29. On plant improvements (Page 3-1 to 3-5 of the IPE submittal):

    A number of plant improvements pertinent to human performance were identified from the original PRA or the revision. Some have been implemented and some are under evaluation. The information presented in the submittal is too sparse to make any judgement as to likely effectiveness of the enhancements. For example:

a) "On SSF Operator Action," explain "the MNS PRA results are relatively sensitive to the SSF actions and there is some uncertainty associated with the operator action....efforts are under way to achieve consistent operator response within approximately 10 minutes".

b) On reactor Coolant Pump Restart Criteria, explain "additional procedure guidance to permit pump startup only when SG tubes are covered with a mixture level has been recommended to eliminate this concern. The procedure change process has been initiated to accomplish this enhancement."

c) On nuclear service water RN cross-connect, explain "There is some uncertainty on the probability of successful timely action. It has been suggested that a periodic exercising of these valves (during refueling outages) could enhance the confidence on this recovery action."

Provide more specific information as to exactly how the improvements were identified, what they involve, what systematic processes were used for identifying the best improvement method, the expected impact on human performance, and a schedule for their accomplishment.

30. Explain the basic event "SLOWTHRATE." Why was it stated that the heat transfer from corium to water pool was small? MAAP (the tool used for analyses) uses heat transfer rates close to the critical heat flux of water for the calculation of heat transfer rates from corium to water. What is meant by slow heat transfer rates in this case?

31. You indicate that the McGuire PRA update was completed in January of 1988. Identify any modifications to the PRA to account for plant modifications or procedures changes since January 1988. Discuss your intentions regarding the maintenance of the PRA as a "living" tool.

32. Identify and discuss any outside peer review performed on the PRA. Identify those experts who performed the review and discuss their findings and your responses.

33. Provide a discussion explaining the finite element model used in the containment failure analysis in Appendix G of the PRA.

34. Identify and discuss the contributors to containment isolation failure, i.e. isolation signal failure, valve failures (including containment purge values), degradation of valve seats etc., and the overall probability of isolation failure used in the PRA.

35. Provide frequencies for the release categories of Tables 6.3-1 to 6.3-25.

36. Explain how source terms were evaluated for important release categories. Were they the MAAP-calculated source terms for the

dominating sequence of each release category? In addition, explain how the source terms were obtained for Release Categories 1.02 and 1.04. Did MAAP calculations show any potential for induced SGTRs?

37. On page 2-18 of the submittal it is stated, "Recovery of containment sprays late after core melt is an important mitigating action." Explain what emergency operating procedures are in place for this recovery action. Also discuss how you analyzed the possible negative effects of spray recovery, such as hydrogen burn.

38. Hydrogen combustion is an important phenomena associated with ice condenser containments. Describe the process used to treat the potential for hydrogen detonation and the impact of detonation on the containment and containment systems. For the hydrogen ignitor analysis in Section 5 of the IPE, did the base case (Station Blackout) consider restoration of ac power (which would have energized the glow-plugs) and a subsequent hydrogen burn? Also for the base case, the value given for "NOSPARKLT" (page 6.2-43 of PRA) is 0.75. Thus, 25 percent of the time, random sparks were available to ignite hydrogen after RV failure. What are the implications of resulting hydrogen burn on containment failure? Where in the containment were the sparks occurring? Please provide the information requested in NUREG-1335 (Section 2.2.2.1), i.e., accurate but simple representations of the containment showing the Instrument Tunnel, Reactor Cavity compartment, Loop compartment(s), Annular compartment(s) and Upper compartment with specific identification of potential reactor release points and vent paths indicated. Estimates of compartment free volumes and vent path flow areas should also be provided. Address specifically how this information is used in your assessment of local hydrogen pocketing and detonation.

39. NUREG-1335 recognizes the importance of considering uncertainties in the accident progression and CET quantification. EPRI recommends that sensitivity studies be performed by MAAP users, which could provide qualitative insight into understanding uncertainties. Please specify what specific revision(s) of the MAAP-3.0B code were used for the McGuire PRA. Please address the Gabor Kenton & Associates report prepared for EPRI ("Recommended Sensitivity Analyses for an Individual Plant Examination using MAAP-3.0B). In particular with respect to Appendix A of the report, indicate for each of the 78 indicated parameters:

a) If you used the recommended value

b) If you used a value other than the recommended value, the basis for your choice; or

c) If you did not perform the sensitivity study indicated, provide your reasons for omitting the recommended analyses.

40. Identify and discuss the major phenomenological contributors to early containment failure. Explain the basic reasons why early failure is low, compared with early failure in the Sequoyah NUREG-1150 study.

Likewise, identify and discuss the major phenomenological contributors to late containment failure and explain why the conditional probability of lage containment failure is significantly greater than for Sequoyah.

41. NUREG-1335 (Section 2.2.2.4) requested that licensees submittal should include an assessment of penetration elastomer seal materials and their response to prolonged high temperature. Describe the treatment of elastomer seals in your IPE, and any associated findings, results and conclusions. Of particular significance is the rather unique design of personnel access hatch seals used at McGuire, i.e. inflatable seals. Address the results of testing of inflatable seals reported by M.B. Parks in 0029-5493/91 Elsevier Science Publishers 1991. [Nuclear Engineering & Design, Vol. 131, 1991, pages 175-186]

    The referenced report concludes that the viability of inflatable seals at temperatures in excess of about 350°F is poor. Provide pressure/temperature time histories for the worst conditions in the vicinity of the personnel hatches as calculated by the MAAP code, and discuss their significance in light of the above conclusions regarding seal response to elevated temperatures.

42. NUREG-1335 (Section 2.2.2.6) recognized the importance of availability and survivability of systems and components during severe accidents. Identify those systems/equipment which are assumed to remain operable in post accident environments (i.e., containment sprays, hydrogen mixing systems, etc.) and describe the IPE's treatment of equipment exposed to containment conditions during a degraded core accident and any important insights gleaned from the analysis.

43. Provide a table listing the frequency of the plant damage states and, for each dominant plant damage state, provide the split fraction for each nodal question related to each top node event in the event tree.

44. In light of the SGTR experience at McGuire and the most recent pulled tube specimen testing results, have you utilized the PRA to perform sensitivity analysis for SGTR sequences? Recent tube tests and eddy current critical crack detection failures seem to indicate a potentially higher than expected uncertainity for SGTR probabilities. The current PRA indicates that, although the containment bypass probability is relatively low (2.4%), it is dominated by induced SGTRs. It is possible that the containment failure probability profile could be significantly affected by an increased uncertainity concerning SG tube integrity under steam line failure and post accident SG dryout conditions. We believe that it is important that the PRA be used to develop a perspective of the sensitivity of plant risk to the potential increased uncertainty in SG tube integrity under these accident conditions. Discuss the PRA insights in regard to this concern.

QUESTIONS AND COMMENTS ON GI-105, "INTERFACING SYSTEM LOCA IN PWRS"

Additional information is needed for resolution of GI-105, "Interfacing Systems LOCA at LWRS" at McGuire Unit 1. The following questions and comments apply to the analysis and discussion of ISLOCA in the McGuire Unit 1 PRA.

1. Screening of systems from detailed ISLOCA analysis on the basis of number of normally closed valves in series could result in important ISLOCA pathways being omitted. In particular, all procedures involving operation of pressure isolation valves (PIVs) should be analyzed for:

    a. identification of possible errors of commission involving motor-operated PIVs?
    b. identification of procedurally sanctioned or other defeat of PIV interlocks?
    c. inclusion of appropriate warnings and independent verifications regarding PIV operation?

    Subsidiary non-PIV valves can also be important depending upon system configuration, test, and operating procedures.

2. Have you analyzed all the procedures which involve any PIV operation during transition between modes of operation?

3. To what extent has industry operating experience been factored into the analysis, including human errors? There is little evidence of this in the PRA.

4. Are PIVs leak-tested individually or together as one barrier?

5. Rupture of piping system components is a function of temperature, amount of overpressurization, and component fragility (cf. D.A. Wesley, et al. "Pressure-Dependent Fragilities for Piping Components," NUREG/CR-5603, October, 1990) with pipes generally being the least fragile. There should be less discussion of pipe failures and more analysis of likelier failure locations such as ND heat exchangers (HX). HX failures that are possible due to overpressure include cylindrical (hoop) failure, head buckling, and tubesheet failures. Possible leak areas can be equivalent to a guillotine break of a pipe feeding the HX.

6. Likely failure location identification is important to assess ISLOCA flooding and environmental effects on valves and injection equipment necessary for break isolation and accident recovery. What assumptions were used in the McGuire Unit 1 ISLOCA analysis with respect to availability of valves and pumps following an ISLOCA ?

7. Assuming valves required for break isolation are available following ISLOCA, are they able to operate against full NC pressure?

8. NUREG/CR-5604 (draft) and two companion volumes produced by the NRC GI-105 resolution program have indicated that because ISLOCA frequencies can be higher than estimated by past PRA approaches to the problem, recovery actions can be important to prevention of core melt. The 2E-6 per reactor year core melt frequencies from these studies include contributions from operator failures in detection, diagnosis, isolation, and mitigation actions. However, the McGuire Unit 1 PRA analysis derived ISLOCA frequencies that are so low that subsequent recovery actions could be ignored, in most cases. The two isolation events described are not simulated, but operators receive classroom training on them. The adequacy of recovery training and procedures does not appear thoroughly addressed. Specifically:

a. Are events ISLOCA1DHE and ISLOCA2DHE the only ISLOCA isolation actions operators are trained in?

b. Do EPs specifically address ISLOCAs, if not by that name then some other?

c. During a LOCA how much time is spent by operators on other activities (i.e. verifying startup and operation of various equipments) before ISLOCA diagnosis and isolation activities are allowed by EPs?