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

September 15, 1995

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON McGUIRE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL (TAC NOS. M83639 AND M83640)

Dear Mr. McMeekin:

Based on our ongoing review of the McGuire IPEEE submittal and its associated documentation, we have developed the enclosed request for additional information (RAI). The RAI is related to the external event analyses in the IPEEE, including the seismic analysis, the fire analysis, and the analyses on the effects of high winds, floods, and others. This RAI was developed by our contractor, Energy Research, Inc., and reviewed by the "Senior Review Board" (SRB). The SRB is comprised of the Office of Nuclear Regulatory Research (RES) and the Office of Nuclear Reactor Regulation staff and RES consultants (Sandia National Laboratory) with probabilistic risk assessment expertise for external events.

In order for us to maintain our review schedule, we would appreciate it if you could provide us your response within 60 days from receipt of this letter. For questions concerning our review, please contact me at 301-415-1484.

This requirement affects nine or fewer respondents and therefore it is not subject to the Office of Management and Budget review under P.L. 96-511.

Sincerely,

ORIGINAL SIGNED BY:

Victor Nerses, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosure: Request for Additonal Information

cc w/encl: See next page

DISTRIBUTION

Docket File	OGC
PUBLIC	ACRS (4)
PD22 Reading File	R.Crlenjak,RII
S.Varga	E.Merschoff,RII
J.Zwolinski	

DOCUMENT NAME: G:\MCGUIRE\MCG.01

OFFICE	DRPE/PD22/LA	DRPE/PD22/PM	DRPE/PD22/D	
NAME	L.BERRY	V.NERSES	H.BERKOW	
DATE	9/15/95	9/15/95	9/15/95	
COPY	YES NO	YES NO	YES NO	

OFFICIAL RECORD COPY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 15, 1995

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON McGUIRE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL (TAC NOS. M83639 AND M83640)

Dear Mr. McMeekin:

Based on our ongoing review of the McGuire IPEEE submittal and its associated documentation, we have developed the enclosed request for additional information (RAI). The RAI is related to the external event analyses in the IPEEE, including the seismic analysis, the fire analysis, and the analyses on the effects of high winds, floods, and others. This RAI was developed by our contractor, Energy Research, Inc., and reviewed by the "Senior Review Board" (SRB). The SRB is comprised of the Office of Nuclear Regulatory Research (RES) and the Office of Nuclear Reactor Regulation staff and RES consultants (Sandia National Laboratory) with probabilistic risk assessment expertise for external events.

In order for us to maintain our review schedule, we would appreciate it if you could provide us your response within 60 days from receipt of this letter. For questions concerning our review, please contact me at 301-415-1484.

This requirement affects nine or fewer respondents and therefore it is not subject to the Office of Management and Budget review under P.L. 96-511.

Sincerely,

Victor Nerses

Victor Nerses, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosure: Request for Additonal Information

cc w/encl: See next page

Mr. T. C. McMeekin
Duke Power Company

McGuire Nuclear Station

cc:

A. V. Carr, Esquire
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242-0001

County Manager of Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Mr. J. E. Snyder
Regulatory Compliance Manager
Duke Power Company
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

J. Michael McGarry, III, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Senior Resident Inspector
c/o U. S. Nuclear Regulatory
Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Mr. T. Richard Puryear
Nuclear Technical Services Manager
Westinghouse Electric Corporation
Carolinas District
2709 Water Ridge Parkway, Suite 430
Charlotte, North Carolina 28217

Dr. John M. Barry
Mecklenberg County
Department of Environmental
Protection
700 N. Tryon Street
Charlotte, North Carolina 28202

Mr. Dayne H. Brown, Director
Department of Environmental,
Health and Natural Resources
Division of Radiation Protection
P. O. Box 27687
Raleigh, North Carolina 27611-7687

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of
Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. G. A. Copp
Licensing - EC050
Duke Power Company
526 South Church Street
Charlotte, North Carolina 28242

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, NW. Suite 2900
Atlanta, Georgia 30323

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

REQUEST FOR ADDITIONAL INFORMATION
ON McGUIRE IPEEE SUBMITTAL

I. Seismic

1. Provide a discussion on whether using LLNL hazard curve would lead to differences in the delineation of dominant seismic sequences or provide the core damage frequency estimate obtained for either the 1989 or 1993 LLNL seismic hazard curve.

2. Provide a plot of the plant-level seismic fragility curve and a table of values that completely defines the fragility curve.

3. The human error rates provided in the submittal do not appear to take into account seismic events in that the values used are typical of non-seismic scenarios. Demonstrate that these human error rates are representative of responses during the risk significant seismic events, or provide the effect on risk (core damage frequency, dominant cut sets, the frequency of dominant cut sets) of human error rates appropriate to seismic events, if they are not.

4. Typically ice condenser failures accompany many core melt sequences. However, ice condenser fragilities were not provided in the submittal. NUREG-1407 suggests that screening out the ice condenser would not be appropriate. Provide ice condenser fragility calculations, walkdown notes, and any other documentation related to the condition and seismic capacity of the ice condenser.

5. Provide a more detailed description of the seismic plant walkdowns and seismic-fire interaction walkdowns. Include the following information:

   a. Provide the level of experience, training, and extent of involvement of each walkdown participant.

   b. Provide the basis followed for component screening including the assignment of generic fragilities and the conditions affecting plant-specific fragility calculations.

   c. Describe how the walkdown process and findings addressed passive components such as electrical raceways, cable trays, HVAC ducts, and piping.

6. It appears that the study used two sets of in-structure spectra, the set used in the original PRA and the set developed in the auxiliary building for the IPEEE walkdown. Provide the independent peer review comments with respect to equipment fragilities and the associated floor response spectra employed in the analysis and the IPEEE study's response to the comments. Also, looking at the list of fragilities provided in Table 3-1, provide which set of in-structure spectra was used for each equipment item (e.g., old or new).

Enclosure

7. In developing new floor response spectra, was deconvolution used? If so, please identify the soil profile and properties which were used in the analysis. How do the new spectra impact the results of the fragility analysis that used the older response spectra?

8. Provide a list of block walls whose failures could impact the function of the safety equipment and provide a description of how these block walls were treated, including walkdown screening and fragility assessment.

9. Please provide a more detailed description of the process undertaken for seismic-fire interaction evaluations. This should include: (a) assessing the adequacy of anchorages for the components (i.e., pumps, CO_2 and water tanks) or the impact resulting from the failure of the fire protection system; (b) whether there are "bad actor" or mercoid relays in the fire protection actuation system; and (c) whether proximity to equipment energized at less than 600 V was examined (and if not, please provide the technical basis for excluding 480 V switchgear, etc., given available experience with failures of such equipment).

10. Cabinets, panels, motor control centers, and control boards must be adequately anchored to resist ground motion. The relay and cable attachment points are particularly sensitive to relative motion. The study documentation did not identify anchorages that were inadequate and had to be replaced. It is very unusual for all anchorages to be screened out. The system model does not explicitly include seismic anchorage failure, which is typically a significant contributor to core damage frequency. Provide an analysis and documentation of the methodology, assumptions, calculations, and results used to screen out anchorages.

11. Provide details of the fragility calculations for the following components which either differ from the original 1984 PRA or were added since then: containment spray heat exchanger, residual heat removal heat exchanger, residual heat removal pump, main control boards, auxiliary shutdown panels, motor-driven auxiliary feedwater pump control panels and turbine-driven auxiliary feedwater pump control panels. What in-structure spectra were these analyses based on?

12. Provide a list of components important to decay heat removal functions, their seismic capacities, and the contribution or importance to seismic core damage frequency.

13. All seismic events were treated using the transient event tree in the seismic system model. LOCAs are limited to non-seismic failure to close pressurizer PORVs and functionally dependent inducement of RCP seal leak. Provide the analysis and documentation (methodology, assumptions, calculations, and results) used in the IPEEE of the potential for LOCA initiating events include failure of Reactor Coolant System equipment and pressure boundaries, chatter of relays related to pressurizer PORVs, and collapse of other structures onto the RCS.

14. The list of equipment that was included in the system model and for which fragilities were generated is small relative to other studies. Typical equipment for which fragilities have been developed in other studies and have been included in the system model is as follows: traveling water screens, pressurizer and supports, pressurizer relief and safety valves, polar crane, control rod drive mechanisms, main steam isolation valves, main steam relief valves, condenser dump valves, main steam safety valves, HVAC equipment (including switchgear, cabinets, and ducting), reactor vessel and nozzles, reactor coolant pumps, steam generators and attached piping. Describe and explain the walkdown process and findings and calculations used to screen out these components. In addition, describe and explain the walkdown process and findings related to passive components such as electrical raceways, cable trays, HVAC duct, electrical connections, and piping.

15. The submittal neither includes an estimate of fragilities for the turbine building nor includes them in the system model to determine core melt frequency. This omission of a potentially important interactions may be significant in light of safety related equipment, such as cabling and piping, that is sometimes routed through the turbine building. Provide a list of equipment from the IPE model that is either found in or can be affected by failure of the turbine building.

16. The submittal states that the IPE transient event tree was used for the seismic system model. However, the top event related to the reactor protection system was not included in the seismic event tree. This has the affect of excluding ATWS events. Explain why the seismic IPEEE study excluded the RPS.

17. The fault tree model appears to have omissions which are typically significant to risk. Examples are: boron injection tank isolation valves on the discharge side of the centrifugal charging pumps; seismic induced failure to open or reclose pressurizer PORVs; and valves associated with alignment of high pressure recirculation. Failure of normally closed motor operated valves can occur not only from low mechanical fragilities, but from seismic failures of the power and control functions.

    Other examples of apparently incomplete fault tree modeling are as follows:

    - Secondary side heat removal is modeled only up to the auxiliary feedwater pumps. The discharge side which relies on the availability of a discharge path through the valves isolating the auxiliary feedwater discharge from the steam generators and through main steam power operated relief valves (or through the turbine if MSIVs are still open) are not modeled.

- Model includes an alternate source of water to the auxiliary feedwater system from the nuclear service water system. However, the crucial isolation valves separating service water piping from auxiliary feedwater suction header is not modeled.

- Model includes the steam turbine driven auxiliary feedwater pump. However, the normally closed isolation valves from the main steam tap to the pump are not modeled.

- Model includes loss of nuclear service water suction by fouling of the pond and failure of the intake structure. However, blockage owing to loss of traveling water screen motion or complete collapse is not modeled.

- Independent failures such of component cooling, safety injection start, residual heat removal start, and motor driven auxiliary feedwater trains, normally expected in seismic system models, were not found in this system model

If they were, indeed, modeled in the seismic event tree/fault tree model, describe how they were modeled and accounted for in the cut sets and core melt frequency. If they were not modeled, explain why they are not risk significant.

18. The seismic fault tree models do not appear to include all the failure modes included in the internal events analysis. Provide a description (including methodology, assumptions, calculations, and results) of how these failure events were screened out of the seismic analysis, and identify the risk significance of the screened failure modes. In addition, if not otherwise addressed, please indicate whether a conditional probability cutoff was used (e.g., 10^{-2} per demand), and if so, discuss the technical basis for this cutoff value.

II. <u>Fire</u>

1. The submittal provides a list of areas that were deemed critical based on an initiating event criterion. How the list of critical areas was determined was not explained. An explanation is necessary in order to determine rooms that may have been inappropriately screened out. A general concern is that the combination of using a single scenario to represent an area coupled with using the criterion of screening out areas because the selected scenario's equipment damage occurs at a lower frequency than random equipment damage misses a key point of fire analysis. This point is that fires tend to cause a demand on shutdown systems and disable shutdown equipment in a way that is not obvious unless looked at in detail. Screened out areas, such as switchgear rooms, therefore, may have emerged as important risk contributors if allowed to be carried into the detailed systems analysis. Provide: (1) the initiating events, from Table 3.5-1, Rev.2 of Appendix B of the submittal, assigned to each area; (2) all sources of fire, other than

the worst case source, considered in each area and what rationale was used to screen them out; (3) analysis and documentation to demonstrate the rationale for screening out each screened out area; and (4) justification that areas screened out are, indeed, unimportant risk contributors. Consider in the answer functional dependencies owing to equipment failed by fire.

2. The submittal assigns a "worst case result" scenario to each critical fire area. If the frequency that the worst case scenario causes redundant equipment damage is either less than 10^{-8} or less than the failure probability of similar equipment in the IPE study, then the entire area is screened out. This approach does not consider the cumulative effect that many less severe fires and other scenarios in other locations within an area could be significant to risk. This is particularly important for the control room, cable room, Vital I&C area, and all switchgear rooms. Provide the analysis and documentation that substantiates that the cumulative risk of fires, other than the worst case scenario selected, are not important risk contributors for these rooms.

3. The physical damage for each selected area scenario related to Stage 1, Stage 2, and Stage 3 of the fire event tree is not provided. They can be deduced by comparing Tables 3.5-5, 3.5-6 and 3.5-7. This comparison provides results that are confusing. For example, it appears that a fire initiated in one pump (e.g., nuclear service water) and fails that pump is considered a Stage 2, except that a Stage 2 fire was treated as incapacitating both main feed pumps. Provide a description of the specific equipment assumed to be damaged for each scenario and each critical area at Stage 1, 2 and 3.

4. The fire initiation frequencies used in the study were based on pre-1983 industry-wide data. Substantiate that the risk estimates are not significantly affected by use of more recent industry-wide data and plant specific data.

5. It appears that the fire initiation frequencies were based on selecting a component in an area and estimating the fire frequency of that component. The total fire frequency of the area must be considered in the analysis not just the frequency of an individual component. The accepted methodology to estimate the area fire frequency is to combine the fire frequencies from all sources in an area (stationary and transient). A comparison of the fire initiating event frequencies of the cable and control areas used in this study with that of a more recent database suggests that frequencies are about a factor of two too low in the McGuire study. Provide a description of the development of fire initiation frequencies for each fire area. Provide the effect on

fire initiation frequency of including fires from all sources (stationary and transient) in each critical area. Substantiate that the method and results of the fire initiating event frequency analysis accounts for fires from all sources (stationary and fixed) in an area.

6. The submittal has not included sufficient explanation of cable and cabinet fires, particularly in light of transient combustibles, to rule out LOCA's (especially pressurizer PORV opening) as an initiating event. The study mentions that the ability to close open PORVs exists by removing power. The ability to do so by no means assures that it will be done during a severe fire, and also does not take into account the possibility of a stuck open or improperly reseated PORV. Provide the area by area analysis with documentation of the potential for the occurrence of a LOCA and detailed explanation for why LOCA scenarios were screened out. Include stationary source fires, cabinet fires and transient combustible fires in the analysis.

7. The submittal does not address the effect of transient combustibles on the potential for component damage and on the fire core damage frequency. This could be a serious omission that underestimates risk. Provide an explanation of the treatment of transient combustible fires.

8. That fires initiate in electrical cabinets is well known. The submittal mentions cabinet fires in only a few areas such as 4160 V switchgear, reactor trip switchgear, and auxiliary shutdown panel but all cabinet fires have been screened out. Provide an explanation of the treatment of cabinet initiated fires and explain how all cabinets in the plant have been screened out. Provide an analysis, if available, of the effects of cabinet initiated fires on the fire risk of McGuire.

9. We agree that hot shorts are not a serious problem for McGuire. However, the practice of reducing the dominant fire cut set frequencies (for the control and cable rooms) by a factor of 5 assumes that the hot short is the only way that equipment is damaged. This is invalid. Clearly, equipment becomes non-functional if its power or control cable is damaged, or if its breaker or fuse opens. Because recovery from equipment non-functionality is probabilistic, it can not be dismissed from the analysis as was done. Including shorts to ground, which can cause equipment to change state and require resetting or repositioning, would introduce scenarios that were not considered in the McGuire submittal. The treatment in the study, therefore, may underestimate core damage frequency. Provide a risk assessment of the control and cable rooms that includes the effect of shorts to ground. Discuss the significance of these scenarios as compared to hot shorts to cable.

10. The study's fire detection and suppression analysis is also imbedded in the fire event tree--NUREG/CR-0654 method. The approach used in the study may be a significant factor in the screening out of many of the areas and in the underestimate of core damage frequency. Implementation

of the approach produces detection and suppression probabilities that are unrealistically low in comparison with more recent data. One of the reasons may be the judgmental adjustment of the NUREG/CR-0654 values. Another reason is the multiple independent opportunities for detection and suppression explicitly modeled in the fire event tree. This inherently makes assumptions that may not be realistic. For example, it implicitly assumes that failure of automatic suppression will always be accompanied by a second and third attempt in time to prevent a Stage 3 fire (by either auto-systems or manual). The suppression probabilities provided in Table 3.5-5 are typically 0.1, 0.8, and 0.1, for a product of $8x10^{-3}$. For the control room, the product is $4x10^{-3}$. For the auxiliary feedwater room, the product is $2.4x10^{-4}$, and for the service water and component cooling water rooms the product is $1.2x10^{-3}$.

In addition, detection probabilities are treated separately. There are two opportunities in series to detect the fire. These are typically 0.1 and 0.05 to 0.01, for a product of $5x10^{-3}$ to 10^{-3}. Typical automatic system failure probabilities used in other studies are on the order of 10^{-2}. This includes both detection and suppression.

The accepted method analytically assesses the competition between fire growth and detection/suppression as a function of time. Provide the affect on area screening and risk of using a more realistic treatment with respect to the time of the onset of damage versus realistic capabilities for the response time of suppression.

11. Fire brigade response times were assumed to be 10 minutes for any plant area and the submittal stated that this was verified during the walkdown. How does this time relate to the time required to suppress a fire. How was this 10 minute time used in estimating the parameters of the fire event tree for each area.

12. The analysis of cut sets involving the control room assumes a Stage 2 fire that was sufficient to fail redundant trains of equipment. While this may be the worst case with respect to the ability of the plant to deal with the situation, it may not capture the majority of the risk with respect to total core damage frequency. For example, typical fire scenarios in control rooms involve smoke that is sufficient to force operators to abandon the control room either because of the adverse environment or because control is lost from smoke damage. This category of scenarios, which would include a variety of potential initiators and loss of equipment functionality, would comprise somewhat less severe but far more likely challenges to the operators and shutdown systems. The cumulative effect of less severe challenges could be of higher risk than the single "worst case" challenge when considering both fire induced failures, human errors, and independent failures. Three other concerns with the approach used in the study are as follows:

a. The representation of all control room fires as a loss of nuclear service water provides the incorrect perception that the effects of all fires can be treated by use of the Standby Shutdown Facility.

b. Typically, control room fire procedures require the abandonment well before a Stage 2 fire. Assuming that operators will stay in the control room is counter to procedures and provides an optimistic perspective on the scenarios.

c. The treatment of Control Room fires in the study does not investigate the operators' and plants' ability to control the plant after a fire using the Alternate Shutdown Panel.

For each area that survived the screening and was used in the fire risk estimates (i.e., control room, cable room, auxiliary shutdown panel, Vital I&C area, turbine building), justify that the "worst case" approach produces an accurate estimate of risk for all scenarios and fire locations in the area. Discuss how the study considered fires, such as a Stage 1 fire that leads to abandonment of the control room, which were less severe than the "worst case" but could lead to core damage.

13. The quantification of the main feed pump fire shown on Page 3.5-18 and 3.5-19 of Appendix B of the IPEEE submittal is confusing and may be in error. It appears that the frequencies of main feed pump fire (causing loss of off-site power), common cause failure of both diesels, and loss of turbine driven auxiliary feedwater pump were multiplied to obtain the scenario core melt frequency. This implies that the auxiliary feedwater pump is part of a core melt minimal cut set with failure of both diesels. If both diesel generators fail after a loss of off-site power, however, core melt will occur owing to an RCP seal LOCA and loss of safety injection unless the leakage is low enough for the Standby Shutdown System to compensate. Failure of auxiliary feedwater is irrelevant and its inclusion in the cut set only serves to underestimate the core melt frequency by nearly two orders of magnitude. The IPE study recognized that there is a difference in plant damage states between success and failure of auxiliary feedwater following a station blackout and includes both scenarios.

An alternative, which is in some emergency operating procedures, is to attempt a steam generator blowdown to cool and depressurize the primary system. This, however, is only a temporary measure that would be successful only if makeup flow can be recovered before primary system pressure rises again. Provide a detailed explanation of the quantification of this scenario.

14. One of the Sandia Fire Risk Scoping Study issues is seismically-induced fires. Explain the basis for screening out electrical equipment rated at less than 600V. Explain the basis for screening out all bottles and tanks containing 5 gallons or less of flammable materials, when pilot fires of 3 gallons can cause damage to adjacent equipment.

III. High Winds, Floods, and Others (HFO)

1. Please provide a list of any significant changes with respect to plant design against HFO events since the McGuire PRA was issued.

2. Please provide a summary of the walkdown findings related to HFO events.

3. Provide justification for crediting SSF equipment in response to a tornado (as shown in the cutsets provided in Table D-6 of the McGuire PRA), even though the SSF is not a Class I structure.