September 15, 1995

Mr. M. S. Tuckman
Senior Vice President
Nuclear Generation
Duke Power Company
P. O. Box 1006
Charlotte, NC 28201

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - CATAWBA INIDVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) (TAC NOS. M83605, M83606)

Dear Mr. Tuckman:

Based on our ongoing review of the Catawba Individual Plant Examination of External Events (IPEEE) submittal dated June 21, 1994 and its associated documentation, we have developed the attached 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 effects of high winds, floods, and others. Please provide a response within 60 days of receipt of this letter to enable us to continue our review.

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:

Robert E. Martin, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414

Enclosure:
Request for Additional Information

cc w/enclosure:
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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Project Directorate II-2
Division of Reactor Projects - I/II
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Docket Nos. 50-413,
and 50-414

Enclosure:
Request for Additional Information

cc w/enclosure:
See next page

Mr. W. R. McCollum
Duke Power Company

Catawba Nuclear Station

cc:
Mr. Z. L. Taylor
Regulatory Compliance Manager
Duke Power Company
4800 Concord Road
York, South Carolina 29745

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

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

North Carolina Municipal Power
Agency Number 1
1427 Meadowwood Boulevard
P. O. Box 29513
Raleigh, North Carolina 27626-0513

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

County Manager of York County
York County Courthouse
York, South Carolina 29745

Richard P. Wilson, Esquire
Assistant Attorney General
South Carolina Attorney General's
Office
P. O. Box 11549
Columbia, South Carolina 29211

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

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

North Carolina Electric Membership
Corporation
P. O. Box 27306
Raleigh, North Carolina 27611

Senior Resident Inspector
4830 Concord Road
York, South Carolina 29745

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

Max Batavia, Chief
Bureau of Radiological Health
South Carolina Department of
Health and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Mr. G. A. Copp
Licensing - EC050
Duke Power Company
526 South Church Street
Charlotte, North Carolina 28242-0001

Saluda River Electric
P. O. Box 929
Laurens, South Carolina 29360

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

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

REQUEST FOR ADDITIONAL INFORMATION

ON CATAWBA IPEEE SUBMITTAL

I. Seismic

1. Explain how the seismic IPEEE analysis evaluated the potential for flooding to arise from seismically-initiated failures.

In particular, the NRC staff's review of the Catawba IPE submittal (as documented in a letter dated June 7, 1994, identified turbine building floods to be a significant contributor to CDF due to the location of all 6.9 and 4.16 kV transformers at Elevation 554 of the turbine building basement. (The 1992 Catawba IPE report clarifies that the flooding of concern arises from a large leak in the condenser circulating water system; see Section 3.3.7.3. The 1992 IPE also indicates that both units would lose power to the 4.16 kV essential buses; see Section 8.1.3. Thus, such a scenario could involve a concurrent, dual-unit, station blackout event.)

Neither the EPRI seismic margin assessment of Catawba Unit 2 (EPRI NP-6359) nor the 1992 IPE submittal nor the 1994 IPEEE submittal indicate that the condenser circulating water system was analyzed for seismic fragility. A quick search of a compilation of fragility information from available PRAs performed in the 1980's identified median PGA fragilities of 0.71g for a PWR main condenser and 0.26g for a PWR condenser hotwell; in both cases, failure of supports caused the median fragility to be much lower than for typical piping. This suggests that the fragility of the condenser circulating water system may be relatively modest, although in excess of the SSE.

In addition to the request above, please identify the fragility parameters which were estimated for the condenser circulating water system inside the turbine building (i.e., median fragility and uncertainty parameters). If no such parameters were estimated, please describe the basis for assuming that this non-safety-related system does not fail in a beyond design basis earthquake. Finally, provide the results of any seismic walkdown which was conducted on the condenser circulating water system piping within the turbine building.

(Note further, that although in the internal flooding analysis the SSF would normally be available, in the case of a seismic initiator this would be less likely, due to seismic-related failures of the SSF, which is not a seismically designed structure.)

2. The Catawba IPEEE seismic PRA assigns a value of 0.1 per demand for failure to recover from relay chatter (Table 3-2). This is a generic value, however the approach to such a generic human reliability analysis (HRA) appears to mask potential vulnerabilities due to the role of this recovery value in dominant cut sets. There is no generic basis provided

for this assessment, and in contrast to this generic treatment the actual recovery probability would appear to be scenario, relay, and location specific. Indeed, it must be demonstrated in each case that the relay failure mode is in fact recoverable. Please provide the documentary basis for the estimated human error probability of 0.1 for failure to recover from relay chatter for the dominant cut sets, addressing as appropriate any scenario, relay, or location-specific factors which could affect the assessed recovery value. In addition, address the sensitivity of the results of the seismic IPEEE to the use of scenario, relay, or location-specific factors. Explain how the effects of the initiating earthquake on operator stress levels were reflected in the recovery probabilities and, if they were not reflected, please describe how and to what extent stress would affect the recovery probabilities (i.e., is there a dependence of recovery probabilities on seismic ground motion intensity). Finally, for each of the recovery actions included in the dominant cut sets, please identify the applicable plant procedures which identify the required actions to recover from the specific relay chatter failure modes postulated in the sequence.

3. Please map the cut sets reported in Tables 3-4 and 3-5 into the functional event sequences identified in Section 3.1.5.2 and Figure 3-2, and, as requested in NUREG-1407, for the dominant sequences (using the selection criteria above) provide for each sequence the estimated frequency of core damage and their percentage contribution to overall seismic CDF. In addition, for each functional event sequence, please provide information on the timing of core damage as requested in NUREG-1407, Section C.2.1.

4. Please provide a summary of the walkdown findings in regard to seismically-induced failure of containment isolation.

5. The relays used to perform containment isolation functions in the ESFAS and SSPS systems appear to have sufficiently low capacities that failures of SSPS show up in dominant cut sets. Please assess the impact on containment isolation as a result of SSPS/ESFAS failures, i.e., how failure of SSPS due to unrecovered chatter failure would affect containment isolation, and how these impacts were reflected in the containment performance assessment for seismically-initiated sequences at Catawba. Also, please indicate whether SSPS failure would cause failure of the reactor protection systems and, if so, how the effects of this failure were reflected in the seismic IPEEE analysis. In addition, please identify the component fragilities in Table 3-1 of the IPEEE report which correspond to the panel boards and MCCs which actuate the containment isolation valves; and identify the fragility parameters for the ESFAS system (i.e., are these parameters identical to the SSPS system as implied in the IPEEE submittal report).

6. Please provide a plot of the plant-level seismic fragility curve and a table of values that completely defines the fragility curve.

7. 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).

   d. Discuss the extent to which the walkdowns evaluated the configuration and condition of the containment internals important to the containment performance assessment (i.e., the ice condenser, the containment sprays, the containment air fan coolers, the air return fans, etc.).

8. Please provide a description of how block walls, whose failures could impact the safety function of the plant systems, were treated in the seismic IPEEE, including walkdown screening and fragility assessment.

9. Please describe the procedure and scope used for the relay chatter evaluation conducted for Catawba Nuclear Station, Units 1 and 2.

10. 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 low seismic ruggedness ("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).

11. What is the relationship between the IPE plant logic analysis (event trees and faults trees) and the seismic IPEEE plant logic analysis? Was the internal plant logic analysis modified to address seismic events (if so, how?), or was an entirely new model developed?

12. On page 3-18 of the IPEEE report it is stated that, "External events were judged to have no significant impact on the containment performance model." This statement is contrary to our understanding regarding the impact of seismic events on containment isolation (unrecovered relay chatter or structural failure of SSPS/ESFAS) hydrogen igniters (due to lack of AC power), and containment heat removal. Please explain the technical bases for the judgment that external events have no significant impact on the containment performance model.

13. Please list the components (or classes of components) and their fragility values (if calculated) for components which were not screened out based on the walkdowns, but which are not included in Table 3-1.

14. Please provide a list of components important to decay heat removal functions, their seismic capacities, and the contribution/importance to seismic core damage frequency.

15. 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.

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, battery room, 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 a fire that was initiated in one Nuclear Service Water pump and fails that pump was considered a Stage 2 fire. However, a Stage 2 fire was treated as incapacitating both main feed pumps. A fire that initiates in the 1ETB switchgear and fails that switchgear was considered a Stage 3 fire. 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 Catawba 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 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 component cooling 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 may be inconsistent with plant procedures and thus may provide 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, and KC short room), 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.

7. The submittal has not included sufficient explanation of cable and cabinet fires, particularly in light of transient combustibles, to rule out LOCAs (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.

8. 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.

9. 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 Catawba.

10. The procedure used to screen out areas because equipment damage frequency is lower than the IPE equipment failure frequency screens out areas in a way that is not clear and may prematurely remove areas. For example, a fire in the Turbine Building area at the main feedwater pumps could cause a plant trip and damage cables from the 6900V switchgear rooms. A larger fire could additionally damage instrument air and SSF standby makeup pumps or cables. In addition, a large quantity of combustible materials is typically in the Turbine Building, but the potential for fire was not mentioned. The study states that maximum damage owing to a turbine building fire would be loss of feedwater, off-site power, instrument air and SSF standby makeup pumps. The turbine building was screened out with the argument that suppression systems exist, and the frequency of damage of this equipment was "believed to be much lower than" the frequency of loss of main feedwater followed by loss of off-site power. As pointed out in Question 2, this approach ignores the cumulative effect on risk of many less severe fires. Provide analysis and documentation that shows how all combustible materials in the Turbine Building were treated.

11. We agree that hot shorts are not a serious problem for Catawba. 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 Catawba 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.

12. 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 0.8, 0.8, and 0.1, for a product of 6×10^{-2}. For the control room, the product was 4×10^{-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, for a product of $5x10^{-3}$. 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.

13. 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.

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. <u>High Winds, Floods, and Others</u>

1. Please provide analysis of water build-up on the roof of plant structures as a result of maximum probable precipitation.

2. Provide a summary of the walkdown findings related to HFOs.

3. Provide justification for crediting SSF equipment in response to a tornado (as shown in the cutsets provided in Table D-5 of the Catawba PRA), even though the SSF is not a Class I structure.