Duke Power Company
Catawba Nuclear Station
4800 Concord Road
York, SC 29745

(803)831-3000

DUKE POWER

June 7, 1993

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 50-414
Review of Catawba Individual Plant Examination Submittal
Response to NRC Questions
(TAC Nos. M74394 and M75395)

Reference: Letter from NRC to M.S. Tuckman, dated April 22, 1993

Gentlemen:

In reply to the reference letter, please find the attached response to the questions contained in the NRC's request for additional information concerning the Individual Plant Examination (IPE) submittal for Catawba Nuclear Station.

If you have any questions, please call L.J. Rudy at (803) 831-3084.

Very truly yours,

D.L. Rehn

D.L. Rehn, Vice President
Catawba Site

LJR/s

Attachment

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A011
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xc (W/Attach):
S.D. Ebneter, Regional Administrator
Region II

R.J. Freudenberger, Senior Resident Inspector

R.E. Martin
ONRR

Duke Power Company
Catawba IPE
Response to NRC Questions

June 1993

Catawba IPE
NRC Questions and Duke Responses

Question 1

Given that the design and operation for the Catawba facility are essentially the same as that of the McGuire facility, provide any major design and operational differences (along with the differences in major assumptions) that are found to impact the frequency of major functional sequences for the above facilities (e.g., TBU sequences).

Response

The major differences in the system and analysis aspects which affect the important sequence frequencies are presented in the following.

Nuclear Service Water/Component Cooling Systems

At McGuire, the Nuclear Service Water (RN) System cools the Component Cooling (KC) System and most of the safety related motor driven pumps (ECCS and auxiliary feedwater pumps). The KC System cools the Reactor Coolant (NC) pump thermal barrier. In Catawba the KC System, which is cooled by the RN System, cools the NC pump thermal barrier and the ECCS and motor driven auxiliary feedwater pumps. Therefore, accident sequences involving loss of NC pump seal cooling and ECCS failure can occur with either the loss of RN System or loss of the KC System at Catawba, while for McGuire only the loss of RN event is of significance.

Although the RN Systems between units are interconnected at both plants, the cross connect is kept isolated at McGuire and open at Catawba.

For McGuire, the Containment Ventilation (RV) System can be used to support critical RN loads in the event of a loss of RN in one unit.

For both plants, the Safe Shutdown System (SSF) provides an alternate system to provide NC pump seal cooling. In the Catawba analysis, the time available for SSF activation for loss of RN and loss of KC events was determined to be on the order of 50 minutes, while the McGuire analysis assumed a 30 minute time interval. Therefore, the operator action for the SSF activation was assessed to be more reliable in the Catawba analysis versus the McGuire analysis. The net effect of these differences is that the core damage frequency from loss of RN/KC events is estimated to be 2.9 E-5/year for Catawba and approximately 1 E-5/year for McGuire.

Loss-of-Offsite Power

Both plants incorporate two station shared auxiliary transformers which can feed the vital AC power loads in one unit from the other unit. In the Catawba analysis, the loss-of-offsite power (LOOP) frequency was determined by considering the industry data involving loss-of-offsite-power to two-unit sites and loss-of-offsite-power to one unit and considering the availability of power from the unit through the shared transformer. In the McGuire analysis, the LOOP

initiator was not distinguished between the 1-unit LOOP and the 2-unit LOOP. This refinement in the initiator treatment resulted in the LOOP initiated sequence to be 1.2 E-6/year at Catawba versus 1.1 E-5/year at McGuire.

The differences in the loss of RN/KC sequence and LOOP sequences largely account for the difference in the TQsU functional sequence (3.3 E-5/yr at Catawba versus 1.8 E-5/year at McGuire).

Recovery of Main Feedwater

The main feedwater (CF) system would be available following a plant trip not caused by a loss of main feedwater. The recovery probability of main feedwater is estimated based on plant operational data. This recovery probability was estimated to be 0.95 for Catawba and 0.8 for McGuire.

This difference is an additional factor influencing the difference in the TBU functional sequence, which involves events such as the loss of CF event, loss of RN/KC events, and loop events.

6900/4160-V Auxiliary Power Transformer

Another difference in the two plants is the location of the 6900/4160-volt transformers. At Catawba, these transformers are located in the basement of the turbine building. This makes them susceptible to large turbine building flooding events. At McGuire, these transformers are located outside the turbine building. Although flooding of these transformers does not lead directly to a core damage accident, it would result in a failure of offsite power for an extended period of time. This difference accounts for the TB flooding sequence at Catawba and not at McGuire.

FWST Level Transmitters

The refueling water storage tank (FWST) at McGuire utilizes a 3-channel level instrument system, while the Catawba configuration is a 4-channel system. The Catawba FWST level instrument system is assessed to be less susceptible to common-mode failure. This difference is responsible for the difference in the sump recirculation failure for the LOCA sequences, and accompanying functional sequences.

Question 2

Provide the following information related to the Catawba plant walk downs performed as part of the IPE:

A. Types of walk downs performed
B. Objective and scope of each walk down
C. A brief discussion of the process used to integrate findings into the plant model (for further modeling or for deciding plant fixes)

Response

There were three types of walk downs performed during the Catawba IPE study. The first type was conducted during the system familiarization phase of the project. These walk downs were carried out by the system analysts, with assistance from personnel from the plant performance and engineering groups. The purpose of this walk down was to gain an over all idea of the configuration of the system and how it was operated.

The second type of walk down was to gain insights on specific questions which arose during the fault tree development and solution phase of the project. These types of reviews were used to determine information such as the accessibility of equipment, the potential for flooding, and the potential for recovery actions. Findings from these walk downs were incorporated into the draft versions of the PRA models.

The third type of walk down was conducted during the independent review phase of the project. These walk downs were used to resolve questions raised by station personnel concerning modeling assumptions. Any findings of this review were incorporated into the final plant models. Also, potential plant enhancements were sometimes explored during these walk downs.

Question 3

Provide additional information related to the peer review process conducted for the Catawba IPE:

A. Briefly discuss the personnel involved in the peer review
B. Activities conducted (e.g., areas of spot checks and audit calculations)
C. Tools used (e.g., procedures and checklists)
D. A summary of major findings made during the peer review activity and a dispositioning of these findings

Response

The PRA was reviewed by independent reviewers from the Catawba site. The review began with kickoff meetings with the individual teams. During these meetings, the objectives of the review were discussed along with some PRA basics and a brief discussion of the section of the PRA. Each review team member was also provided a copy of the PRA section to be reviewed. After the individual team members had reviewed the PRA section, the team met again to discuss comments and provide feedback. Comments were generally provided as marked up copies of the PRA and IPE reports, or as notes taken during the meetings. The comments from the independent reviewers were resolved by the individual analyst responsible from the section being reviewed.

The reviewers utilized their knowledge of the present operating plant to provide feedback concerning the information provided in the PRA reports related to their area of expertise. The tools used by the review teams varied, depending on the section being reviewed. For example, the operations and operations training personnel reviewed the human factors sections and provided feedback on the timing of the required actions, the training and knowledge of the crews, and the clarity of the procedures for the action considered. System models were reviewed by the "expert team" assigned to that system. This "expert team" consisted of plant personnel from the performance, operations,

maintenance and engineering groups. These teams relied on drawings, maintenance procedures, surveillance requirements, and their general knowledge of the plant.

Presentations were also made to the Catawba station management personnel concerning the results, conclusions and proposed plant enhancements. This review and dialogue facilitated the formulation and endorsements of plant enhancements discussed in Section 6 of the IPE Submittal Report.

Appendix A of this response includes the personnel who reviewed each section of the Catawba PRA and a brief discussion of the most significant comments.

Question 4

Provide the version and the date of the CAFTA code used for the Catawba IPE/PRA. Is the same version of the code used for the Oconee and McGuire facilities? If not, please discuss any major differences (among versions) which could affect the quantification of core damage frequency. Your discussion should include the modeling of train level dependencies (for example, treatment of circular logic loops).

Response

The version of CAFTA used for the Catawba, McGuire and Oconee PRA studies was version 2.0d, April 1989. This version of CAFTA identifies any circular logic within the fault trees so that circular logic can be corrected prior to the solving process. For example, loss of offsite power sequences at Catawba create circular logic because the Emergency Diesel Generator System requires the Nuclear Service Water System for cooling, and the Nuclear Service Water System requires the Diesel for power. This circular logic was corrected by deleting the calls from the Diesel Generator tree to the Nuclear Service Water System tree.

Question 5

The staff notes that one of the four actions taken as part of the IPE activity includes improvements to the Catawba training simulator. Please concisely discuss these improvements and the extent to which they refocused training.

Response

The enhancement related to the simulator concerned improvements in training performed on the simulator, not actual improvements to the simulator. This was an enhancement from the original Catawba PRA performed earlier, and was related to the periodic training that operators receive on the simulator. Most of the PRA accidents have always been covered well by the operator training program. However, the PRA study did identify several specific sequences which were added as exercise guides to the bank of simulator training scenarios. The exercise guides provide important information for the simulator instructors, such as the objectives of the scenario, the initial conditions, the failures to be modeled, and the expected operator response. PRA team members worked with the simulator instructors to develop these guides. The PRA sequences added included the following sequences;

- Loss of Main Feedwater
- Excessive Main Feedwater
- Interfacing Systems LOCA
- Failure of ESFAS to Automatically Actuate
- Five Small Break LOCA Sequences
- Two SGTR Sequences
- Turbine Building Flood
- Loss of the Nuclear Service Water System
- Loss of the Component Cooling System

Question 6

DPC has decided that certain plant improvements considered as part of the turbine flood protection are not cost effective. Describe the analysis performed, the decision process, and the results related to disposition of this issue.

Response

As the result of the original Catawba PRA performed earlier, a design study (CNDS-0076) was performed to evaluate the feasibility of several methods to address the turbine building flood concern. The study reviewed the following options;

- moving all the power equipment to a location not susceptible to flooding,
- providing a flood wall around the equipment,
- relocating or changing the operator type for the isolation valves,
- increasing the sump pump capacity.

The design study concluded that the only marginally cost beneficial fix was a 4.5 foot wall around the 6900/4160-volt transformers. Further review of the cost estimate for this modification revealed that it did not consider the detrimental effects the wall would have on the normal operation of the plant. It was also determined that the proposed wall did not protect all the vulnerable equipment in the turbine building basement. Therefore, it was concluded that there was no hardware related enhancement that could be justified through cost benefit analysis. Instead, it was decided to put additional emphasis on improving the likelihood of the recovery actions, i.e., utilizing the SSF as a backup for RCP seal cooling and secondary side heat removal.

Question 7

Appendix G to the McGuire IPE indicates that the containment failure pressure distribution corresponds to a log normally distributed probability function with a median pressure value of 77 psig and a mean pressure value of 76.71 psig. Although the information presented in Appendix G of the Catawba IPE seems to indicate that the Catawba containment structure is identical to McGuire in shell thickness (3/4") and both structures are subject to the same analytical process, the Catawba structure is stronger (i.e., the containment failure pressure distribution corresponds to a log normally distributed probability function with a median pressure value of 84.5 psig and a mean value of 83.93

psig). Please identify and discuss those differences between the two containment structures that result in the increased containment strength for Catawba.

Response

The difference in the mean failure pressures is due to the different grades of steel used in the construction of the containments. Statistical averages from actual material properties were used in the structural analyses.

McGuire is ASME SA-516 grade 60 and Catawba is SA-516 grade 70 steel.

Question 8

The Catawba analysis indicates that the conditional probability of early containment failure (0.005) is significantly reduced in comparison to the early containment probability (.020) for McGuire. Please identify and discuss those factors which allow for a significant reduction in the probability of early containment failure for Catawba. The discussion should also address the reasons for a slightly increased probability of late containment failure for Catawba.

Response

Early containment failures are dominated by failures due to the combustion of hydrogen. Sequences that contribute to failures from hydrogen combustion are those in which the igniters are unavailable during the period of core degradation, typically those with no ac power. The frequency of the loss of offsite power initiator is significantly lower in the Catawba PRA than in the McGuire PRA. In the Catawba LOOP analysis, explicit consideration of 2 unit versus 1 unit LOOP events resulted in a lower frequency than in the McGuire analysis, where such a refinement was not made. Since these sequences usually proceed to core melt only if the onsite emergency power fails, this is an important initiator for plant damage states in which the igniters are not available. This results in a lower proportion of the core melt frequency in plant damage states with no igniters, which in turn leads to a lower conditional probability of early containment failure at Catawba. The difference in the conditional probability of early containment failure is due to the frequencies of the contributing PDSs and not due to any difference in containment performance. Given all the considerations that go into the analyses, the McGuire and Catawba results are not viewed as significantly different.

Late containment failures are dominated by a long term overpressurization from steam generation in the absence of containment spray. Plant damage state (PDS) 19DI is the dominant contributor to the late containment failure frequency for Catawba and it's frequency is a larger fraction of the total core melt frequency than was the case for the dominant late containment failure sequences for McGuire. The slight difference in the conditional probability of late containment failure is due to the frequencies of the contributing PDSs and not due to any difference in containment performance. The table below summarizes some of this PDS information.

PDS	% OF CMF	% OF LCF FREQ.
CNS-19DI	37	90
MNS-22CI	28	47
MNS-7PI	23	26

Question 9

The probability of bypass failure of the containment was determined to be 0.024 at McGuire, and it was indicated that bypass failure was dominated by induced steam generator tube ruptures (ISGTR) (over 90% of the bypass frequency is due to ISGTR). The Catawba IPE indicates the probability of bypass failure is 0.002 and that it is dominated by ISLOCA. It is our understanding that the steam generators at both McGuire Unit 1 (the unit analyzed for the McGuire IPE) and Catawba Unit 1 (the unit analyzed for the Catawba IPE) are scheduled for replacement in the near future, because of severe tube cracking. Please identify and discuss the differences between McGuire and Catawba which provide such widely different insights into the containment bypass failure mode and the significance of these differences.

Response

The ISLOCA and induced tube ruptures contribute to the containment bypass frequency. The induced tube rupture analyses are driving the higher conditional probability in the McGuire results, and these considerations are addressed here.

There are several factors which contribute to the difference in the induced tube rupture contribution to the containment bypass frequency results between McGuire and Catawba. The first is a difference in the CET quantification for determining the occurrence of induced tube ruptures. The CET quantification for McGuire assumed a high likelihood, 0.99, that the operators would start the reactor coolant pumps as instructed by the inadequate core cooling emergency operating procedure. This value is contained in PRA Section 6.2, quantification of the basic event NCONBYOPS. This quantification leads to a high probability of having an induced tube rupture for those plant damage states (PDSs) in which power is available to operate the pumps and secondary side heat removal is not available. We concluded as a result of the McGuire IPE, that operation of the reactor coolant pumps during core degradation with a dry steam generator had a negative influence on plant risk. The emergency operating procedures are being rewritten at McGuire and Catawba. The revised procedures will not call for reactor coolant pump operation if core temperatures are elevated and the steam generators are dry. This change was prompted by the McGuire IPE analysis results. In anticipation of this change, we analyzed Catawba assuming that the reactor coolant pumps would not be operated under these conditions. With this assumption, the quantification for event NCONBYOPS became 0.001. Without forced circulation due to reactor coolant pump operation, the potential for an induced tube rupture is greatly reduced. This quantification change is the dominant reason for the observed difference in the two results.

Another important influence is the frequency of PDSs which contribute to induced tube ruptures due to forced circulation. In the McGuire PRA, PDS 14DI, auxiliary feedwater pump room flood, resulted in a high pressure core melt with power available to the reactor coolant pumps and no secondary side heat removal. The relatively high frequency made this a dominant contributor, 81%, to the internal induced tube rupture frequency in the IPE. There is no similarly important PDS in the Catawba results. If the McGuire CET is quantified as in the Catawba analysis, PDS 14DI and others like it would play a much less important role in the induced tube rupture analysis and significantly reduce the induced tube rupture frequency.

As in the early containment failure case, loss of offsite power (LOOP) initiators also are important contributors to the induced tube rupture results. In these sequences transportation of the hot gases generated in the core to the steam generator tubes occurs due to natural circulation. The CET quantification is the same for the McGuire and Catawba analyses for the natural circulation case. However, the higher LOOP initiator frequency does contribute to a higher frequency of containment bypass.

These factors which increased the induced tube rupture frequency in the McGuire analysis caused the containment bypass frequency to be higher for McGuire even though the ISLOCA frequency is about an order of magnitude higher at Catawba.

Question 10

In response to our request for additional information on the McGuire IPE, DPC indicated that a strategy for restoring hydrogen igniters in small groups, following loss of critical AC power events, is being considered for implementation in the accident management guidance. Please indicate whether this strategy is also being considered for Catawba, and discuss the reasoning for your action with regard [to] this potential strategy.

Response

The general strategy of restoring power to the igniters in small groups following loss of all ac power events is equally applicable to both plants. The implementation could vary at the plants to the extent that the strategy will be very dependent on exactly which igniters are electrically connected to the same breaker. When energizing a particular circuit, all igniters on that circuit will come on. If too broad an area is covered by a group, the burns may not be localized.

Conceptually, the strategy is to burn the hydrogen through a series of small deflagrations, e.g. one compartment at a time. The pressure response due to the individual burns would be mild compared to any event which consumed all of the containment hydrogen at once.

No strategy has yet been developed. It is not clear the necessary analytical tools exist to allow development of the strategy with a high degree of confidence. A code such as HECTR could be used, but HECTR does have limitations which would [illegible] some degree the confidence that the strategy would work as intended. We do plan to perform some HECTR analyses to evaluate the possibility of developing an effective strategy.

Duke Power Company
Catawba IPE
Response to NRC Questions

Appendix A

June 1993

Summary of Catawba IPE Peer Review

Review Subject	Personnel Involved in Review	Major Comments
General Overview of Results and Recommendations (IPE Submittal Report, and Section 8 of the PRA)	M. S. Tuckman Station Vice President E. M. Geddie Superintendent of Operations R. C. Futrell Compliance Manager J. S. Forbes Engineering Manager T. E. Crawford Systems Engineering Manager W. R. McCollum Station Manager J. D. Wylie Training Manager C. W. Boyd Mechanical/Nuclear Engineering Manager E. W. Fritz Systems Engineering Supervisor D. L. Ward Mechanical Engineering Supervisor S. W. Brown Mechanical Engineering Supervisor S. R. Frye Operations Support Manager T. P. Harrall Safety Assurance Manager	This management review group reviewed the results, conclusions and recommendations of the IPE study. Their comments generally involved questions concerning the methods and assumptions, recommendations for who to contact for further information, and helpful suggestions for proposed enhancements.

Summary of Catawba IPE Peer Review

Review Subject	Personnel Involved in Review	Major Comments
Floods (Section 3.3 of the PRA)	J. J. Mackay Civil Engineering K. L. Evans Mechanical/Nuclear Engineering K. R. Caraway Electrical Engineering Supervisor J. E. Herrington Electrical Engineering	The draft PRA indicated that after ten minutes, the valves necessary to isolate the flood would be submerged. However, comments from station personnel indicated that the transformers would already be submerged regardless of the isolation attempts. As a result, all references to a ten minute period was deleted. Station personnel also determined that some important equipment was outside the proposed flood wall. This made the wall a less desirable proposal for addressing the Turbine Building flood concern.

Review Subject	Personnel Involved in Review	Major Comments
Human Reliability (Section 5 of the PRA)	J. E. Teofilak Operations Procedures G. B. Ice Operations Training S. R. Frye Operations Support Manager	Several comments from the independent reviewers concerned the time required to accomplish operator actions or the likelihood of latent human errors. An example of this type comment concerned the time assumed in the PRA analysis for the operators to manually scram the reactor during an ATWS event. The PRA had assumed a response time of 5 to 10 seconds. The reviewers felt that more time would be needed for the operators to respond. This comment resulted in a small increase in the ATWS initiated core damage frequency. Another comment concerned the PRA assumption that a latent human error resulting in a containment isolation failure would be discovered within a short time because of the reduced need for air releases to control normal containment pressure. The reviewers felt that this error would be discovered in this manner but it might require several days. This change resulted in a significant increase in the containment isolation failure probability for sequences which have AC power available.

Review Subject	Personnel Involved in Review	Major Comments
ECCS (Appendices A.1, A.2, A.3 of the PRA)	J. M. Sawyer Safety Analysis R. Menichelli Mechanical/Nuclear Engineering J. A. Kammer Systems Engineering J. G. Torre Systems Engineering	A major comment concerned the length of time that the charging pumps could run without forced cooling to the pumps. The PRA had assumed that the charging pumps would fail fifteen minutes following a failure of the Component Cooling Water System and result in a loss of reactor coolant pump seal cooling and eventually a seal LOCA. Plant personnel were aware of an incident which occurred prior to plant startup where cooling was lost to the charging pumps for more than 30 minutes but resulted in no damage to the charging pumps. This change in the failure time of the charging pumps allows more time for the operators to go to the SSF and start reactor coolant pump seal cooling using the Standby Makeup Pump. This significantly improves the likelihood that the operators will successfully perform this action to prevent a reactor coolant pump seal LOCA following both the failure of component cooling(T10), and failure of nuclear service water(T9).

Review Subject	Personnel Involved in Review	Major Comments
CA System (Appendix A.5 of the PRA)	W. R. Tomlinson Component Engineering M. L. Edmunds Component Engineering G. F. Purvis Systems Engineering P. W. Barrett Mechanical/Nuclear Engineering B. M. Graves Operations H. J. Nicholson Nuclear Services	Comments indicated that failure of the lube oil cooler would not result in failure of the CA turbine driven pump. The model was changed to reflect this comment. Another comment indicated the internals of valve 1CA129 had been removed. The failure mode of this valve transferring closed was removed from the fault tree and deleted from the list of cutsets.

Review Subject	Personnel Involved in Review	Major Comments
RN System (Appendix A.6 of the PRA)	B. G. Felker System Engineering J. F. McKeown System Engineering H. D. Mason Component Engineering D. B. Thompson Operations K. L. Bishop Component Engineering L. J. Benjamin Operations R. C. Bucy, Mechanical Engineering S. W. Brown, Mechanical Engineering Supervisor	Several comments concerned normal valve configurations, or operation of valves during accidents. These comments resulted in several changes to the RN system fault tree. However, they had a minor impact of the results of the fault tree solution.

Review Subject	Personnel Involved in Review	Major Comments
KC System (Appendix A.7 of the PRA)	M. J. LaForrest Component Engineering H. J. Nicholson Nuclear Services C. M. Sahms System Engineering B. S. Dycus Component Engineering D. B. Thompson Operations R. F. Flowers Chemistry W. B. Hallman, Mechanical Engineering R. C. Bucy, Mechanical Engineering	One of the most significant was the PRA assumption that two Component Cooling Water System pumps were required to prevent a Loss of Component Cooling Water initiated event (T10). Station personnel actually tested the pumps in this configuration and demonstrated that a single pump could provide sufficient flow to cool the plant loads. This change in success criteria had a significant affect on the T10 initiator frequency.

Review Subject	Personnel Involved in Review	Major Comments
VI System (Appendix A.13 of the PRA)	M. G. Osteen Mechanical/Nuclear Engineering D. C. Ashburn Electrical Engineering H. J. Nicholson Nuclear Services T. E. Gaye Operations C. B. Cauthen Component Engineering W. C. Adams Mechanical/Nuclear Engineering V. D. King Systems Engineering J. R.Wallace Instrumentation & Electrical	A significant comment concerned the success criteria for the Instrument Air System. The PRA had assumed the configuration and air demand described in the system description. Plant personnel responsible for the system indicated that leaks in the system had increased the air demand to the point that an additional compressor was required during a significant part of the year. Although the failure of instrument air is still not a significant contributor to risk, this change in success criteria resulted in an order of magnitude increase in the T12 initiator frequency.

Review Subject	Personnel Involved in Review	Major Comments
Safe Shutdown System (Appendix A.18 of the PRA)	E. W. Fritz, System Engineering Supervisor D. Davies, System Engineering F. Poley, Electrical Engineering J. Kammer, System Engineering	The SSF system team expressed concern over the treatment of some manual valves in the SSF system. Kerotest manual plug type valves have experienced very few failures at Catawba. A search of the INPO NPRDS data base did not identify any failures of the type described in the fault tree model. Therefore, the probability of failure of these valves was decreased by a factor of ten below the typical generic failure rate.
HVAC (Appendix A.19 of the PRA)	S. B. Putnam System Engineering J. A. Kammer System Engineering	The reviewers express concern over the YM system unavailability values. They believed that the operators would have significant time to correct YM related problems. The unavailability value for YM was decreased to address this concern.

Summary of Catawba IPE Peer Review

Review Subject	Personnel Involved in Review	Major Comments
Transient, LOCA, SGTR, and ATWS Analysis (Sections 2.2, 2.3, 2.4 and 2.5 of the PRA)	G. B. Swindlehurst Safety Analysis Supervisor	No Significant Comments
ISLOCA (Section 2.6 of the PRA)	J. E. Teofilak Operations Procedures G. B. Ice Operations Training	No Significant Comments
Seismic (Section 3.2 of the PRA)	D. R. Kulla Civil Engineering Supervisor W. B. Shoemaker Civil Engineering	No Significant Comments
Tornado (Section 3.4 of the PRA)	D. R. Kulla Civil Engineering Supervisor	No Significant Comments
NC Pressure Control System (Appendix A.4 of the PRA)	Jesse Ray Mechanical/Nuclear Engineering	No Significant Comments
NS System (Appendix A.8 of the PRA)	W. Brown System Engineering	No Significant Comments
Containment Isolation System (Appendix A.9 of the PRA)	G. B. Ice Operations Training	No Significant Comments
Power Conversion Systems (Appendix A.10 of the PRA)	P. W. Hallman Electrical Engineering T. E. Cook Component Engineering	No Significant Comments

Review Subject	Personnel Involved in Review	Major Comments
Vital I&C Power Systems (Appendix A.11 of the PRA)	R. L. Herring System Engineering	No Significant Comments
Containment Air Return and Hydrogen Skimmer System (Appendix A.14 of the PRA)	S. S. Davidson I & E J. M. Yandle Mechanical Engineering	No Significant Comments
Hydrogen Mitigation System (Appendix A.15 of the PRA)	L. R. Wilson Electrical Engineering	No Significant Comments
Essential Aux. Power System (Appendix A.16 of the PRA)	R. L. Herring Systems Engineering	No Significant Comments
Diesel Generator and Load Sequencer (Appendix A.17 of the PRA)	R. L Herring Systems Engineering D. L. Sweat Electrical Engineering F. C. Poley Electrical Engineering R. A. Kayler Systems Engineering W. W. Gallman Nuclear Services W. D. Green Component Engineering E. J. Haack Operations	No Significant Comments

Summary of Catawba IPE Peer Review

Review Subject	Personnel Involved in Review	Major Comments
Reactor Protection and ESFAS (Appendices A.20 and A.12 of the PRA)	J. C. McCart Electrical Engineering	No Significant Comments