FEB 18 1993

Docket Nos. 50-413
and 50-414

Mr. M. S. Tuckman
Vice President - Catawba Site
Duke Power Company
4800 Concord Road
York, South Carolina 29745

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Dear Mr. Tuckman:

SUBJECT: NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M85306 AND M85307)

The Commission has forwarded the enclosed "Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Proposed No significant Hazards Consideration Determination, and Opportunity for Hearing" to the Office of the Federal Register for publication.

This notice relates to your December 15, 1992, application as revised February 5, 1993, to revise the Technical Specifications (TS) as required for the operation of Catawba Unit 2 Cycle 6 after the partial reload of the reactor core with 76 fresh fuel assemblies supplied by the Babcock & Wilcox Fuel Company. The remaining 117 assemblies are Westinghouse supplied Optimized Fuel Assemblies. The proposed TS changes reflect the application of core analysis methodology developed by the licensee and previously approved for the similar reload of Catawba Unit 1 and other changes as discussed in the Notice.

Sincerely,

/S/

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Notice

cc w/enclosure:
See next page

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

February 18, 1993

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and 50-414

Mr. M. S. Tuckman
Vice President - Catawba Site
Duke Power Company
4800 Concord Road
York, South Carolina 29745

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Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Notice

cc w/enclosure:
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UNITED STATES NUCLEAR REGULATORY COMMISSION

DUKE POWER COMPANY

DOCKET NOS. 50-413 AND 50-414

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. NPF-35 and NPF-52 issued to the Duke Power Company (the licensee) for operation of the Catawba Nuclear Station, Units 1 and 2, located in York County, South Carolina.

The proposed amendments would revise the Technical Specifications (TS) as required for the operation of Catawba Unit 2 Cycle 6 after the partial reload of the reactor core with 76 fresh fuel assemblies supplied by the Babcock & Wilcox (B&W) Fuel Company. The remaining 117 assemblies are Westinghouse supplied Optimized Fuel Assemblies (OFA). The proposed TS changes reflect the application of core analysis methodology developed by the licensee and previously approved for the similar reloads of Catawba Unit 1. Changes were proposed to the Safety Limits (TS 2.1 and 2.2) and the Power Distribution Limits (TS 3/4.2.1, 3/4.2.2, 3/4.2.3, 3/4.2.4, and 3/4.2.5) based on using the new licensee analysis methods, a different critical heat flux (CHF), and a new thermal design DNBR (departure from nucleate boiling ratio) limit of 1.55.

The specifications on Catawba Units 1 and 2 TS pages are applicable to both units, with a few exceptions, since the two units are identical in many respects. One of these exceptions involves the transition from fuel manufactured by Westinghouse to fuel manufactured by the B&W Fuel Company (BWFC) combined with a transition in analysis methodology to B&W and Duke Power Company (DPC) methodology. As these changes were first introduced into the Catawba Unit 1 plant, separate TS pages were generated for Units 1 and 2. The changes for Unit 1 in Cycles 6 and 7 reflected the methodology change and a mixed core of BWFC and Westinghouse manufactured fuel while separate pages for Unit 2 continued to reflect the Unit's reliance on Westinghouse methodology and fuel. A similar transition for Unit 2, beginning in its Cycle 6, necessitates similar changes to its TS pages. This is accomplished by deleting the previous pages dedicated to Unit 2 and making the previous pages dedicated to Unit 1 again applicable to both units. Thus, the changes to Unit 1 TS related to the fuel and methodology changes are administrative only, to reflect page renumbering and applicability to both units.

The licensee also proposed TS changes to remove the power range neutron flux negative rate reactor trip (TS 3/4.3.1, Tables 3.3-1, 3.3-2, and 4.3-1); to increase the low steam line pressure setpoint (Table 3.3-4); to increase feedwater isolation and steam line isolation response times (Table 3.3-5); to increase pressurizer safety valve lift setpoint tolerance (TS 3.4.4.2.1); to remove steam line pressure dynamic compensation (TS 3/4.3.2); and to increase main steam line isolation valve stroke time (TS 3/4.7.1.4).

In addition, the licensee proposed TS changes to reduce the flowrate limit for the reactor makeup water pump for Mode 5 (TS 3.3.3.11 and TS 3.3.3.12); to revise the stroke times of valves related to containment isolation (Tables 3.6-2a and 3.6-2b); and to add NRC-approved topical report DPC-NE-1004A to the list of analytical methods used to determine core operating limits (TS 6.9.1.9).

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

POWER DISTRIBUTION AND SAFETY LIMITS

Catawba Unit 1 Cycle 6 was the first [Catawba] Nuclear Station [reload] for which B&W Fuel Company (BWFC) supplied the reload fuel. The Catawba Unit 1, Cycle 6 Reload Report presented an evaluation that concluded the core reload using Mark-BW fuel would not adversely impact the safety of the plant. The Catawba Unit 1, Cycle 7 report was similar, but reflected that Duke Power performed the analyses in support of the operation of Cycle 7 rather than BWFC. This reload for Catawba Unit 2, Cycle 6 is a compilation of the changes made for Unit 1 during Cycles 6 and 7 in that it justifies the use of Mark-BW fuel using Duke Power analysis.

The Catawba Unit 2, Cycle 6 Reload Safety Evaluation Report presents an evaluation which demonstrates that the core reload using Mark-BW fuel will not adversely impact the safety of the plant. During Cycle 6, the core will contain 76 fresh fuel assemblies supplied by B&W and 117 Westinghouse supplied Optimized Fuel Assemblies (OFA).

The changes to the Safety Limit and Power Distribution Technical Specifications presented in Section 8 of the Reload Report represent the application of previously approved methodology to Catawba Unit 2. The changes to remove the power range neutron flux negative rate reactor trip, increase the low steam line pressure setpoint, increase feedwater isolation response time, increase steam line isolation response time, increase pressurizer safety valve lift setpoint tolerance, remove steam line pressure dynamic compensation, ... and increase main steam line isolation valve stroke time reflect the use of Duke analysis, and have already been approved for Catawba Unit 1. The changes described above include the deletion of references to specific units on individual Technical Specification pages, and delete pages which were previously for Unit 2 only. The implementation of unit specific references became necessary due to the transition from Westinghouse to B&W supplied fuel during Unit 1 Cycle 6 and for the Unit 1 Cycle 7 Reload due to the transition to Duke analysis methodology. The analysis which made the changes necessary in the Unit 1 reload submittal is generic, and as described in the technical justification, is equally applicable to both McGuire and Catawba units.

A LOCA evaluation for operation of Catawba Nuclear Station with Mark-BW fuel has been completed (BAW 10174, Mark-BW Reload LOCA Analysis for the Catawba and McGuire Units). Operation of the station while in transition from Westinghouse supplied OFA fuel to B&W supplied Mark-BW fuel is also justified in this topical.

BAW-10174 demonstrates that Catawba Nuclear Station continues to meet the criteria of 10 CFR 50.46 when operated with Mark-BW fuel. Large Break LOCA calculations completed consistent with an approved evaluation model (BAW-10168P and revisions) demonstrate compliance with 10 CFR 50.46 for breaks up to and including the double ended severance of the largest primary coolant pipe. The small break LOCA calculations used to license the plant during previous fuel cycles are shown to be bounding with respect to the new fuel design. This demonstrates that the plant meets 10 CFR 50.46 criteria when the core is loaded with Mark-BW fuel.

During the transition from Westinghouse OFA fuel to Mark-BW fuel, both types of fuel assemblies will reside in the core for several fuel cycles. Appendix A to BAW-10174 demonstrates that results presented above apply to the Mark-BW fuel in the transition core, and that insertion of the Mark-BW fuel will not have an adverse impact on the cooling of the Westinghouse fuel assemblies.

Duke Power Company's Topical Reports DPC-NE-3000, DPC-NE-3001-PA, and DPC-NE-2004-PA provide evaluations and analyses for non-LOCA transients which are applicable to Catawba. The scope of these analyses includes all events specified by sections 15.1-15.6 of Regulatory Guide 1.70 (Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants) and presented in the Final Safety Analysis Report for Catawba. The analysis and evaluations performed for these topicals confirm that operation of Catawba Nuclear Station for reload cycles with Mark-BW fuel will continue to be within the previously reviewed and licensed safety limits.

One of the primary objectives of the Mark-BW replacement fuel is compatibility with the resident Westinghouse fuel assemblies. The description of the Mark-BW fuel design and the thermal-hydraulics and the core physics performance evaluation demonstrate the similarity between the reload fuel and the resident fuel. The extensive testing and analysis summarized in BAW-10173P shows that the Mark-BW fuel design performs, from the standpoint of neutronics and thermal-hydraulics, within the bounds and limiting design criteria applied to the resident Westinghouse fuel for the Catawba plant safety analysis.

Each FSAR accident has been reviewed to determine the effects of Cycle 6 operation and to ensure that the radiological consequences of postulated accidents are within applicable regulatory guidelines, and do not adversely affect the health and safety of the public. The design basis LOCA evaluations assessed the radiological impact of differences between the Mark-BW fuel and Westinghouse OFA fuel fission product core inventories. Also, the dose calculation effects from non-LOCA transients reanalyzed by Duke Power were evaluated using Cycle 6 characteristics. The calculated radiological consequences are all within specified regulatory guidelines and contain significant levels of margin.

The analyses contained in the referenced Topical Reports indicate that the existing design criteria will continue to be met. Therefore, the enclosed TS changes will not increase the probability or consequences of an accident previously evaluated.

As stated in the above discussion, normal operational conditions and all fuel-related transients have been evaluated for the use of Mark-BW fuel at Catawba Nuclear Station. Testing and analysis was also completed to ensure that, from the standpoint of neutronics and thermal-hydraulics, the Mark-BW fuel would perform within the limiting design criteria. Because the Mark-BW fuel performs within the previously licensed safety limits, the possibility of a new or different accident from any previously evaluated is not created.

The reload-related changes to the TSs do not involve a significant reduction in the margin of safety. The calculations and evaluations documented in BAW-10174 show that Catawba will continue to meet the criteria of 10 CFR 50.46 when operated with Mark-BW fuel. The evaluation of non-LOCA transients documented in DPC-NE-3001 also confirms that Catawba will continue to operate within previously reviewed and licensed safety limits. Because of this, the TS changes to support the use of Mark-BW fuel will not involve a significant reduction in the margin of safety.

The technical changes made to Table 2.2-1 reflect the use of the BWCMV CHF correlation and Duke Power's Statistical Core Design methodology with a 1.55 thermal design limit. These changes to Table 2.2-1 will not significantly increase the probability or consequences of an accident previously evaluated. the changes to the K values conservatively bound the allowable operating region, as defined by the new DNBR methodology. It can be concluded that these changes will not create the possibility of any new accident from those previously evaluated. It can also be concluded that since all new TS values are bounded by safety analysis assumptions that this change will not significantly decrease the margin of safety.

DELETION OF NEUTRON FLUX HIGH NEGATIVE RATE TRIP

The removal of the Power Range Neutron Flux High Negative Rate trip will not result in any previously-reviewed accident becoming more probable or more severe. The trip is a response to a pre-existing transient condition and would not initiate any accident. The trip is designed to provide protection from a dropped control rod. However, in the event of a dropped rod, the reactor is assumed to trip on low pressurizer pressure. Therefore, the protection function is retained. The consequences of a dropped rod have been analyzed and found to be within acceptable limits.

Likewise, the removal of this trip will not create a new accident not previously reviewed. The removal of a response to a transient will not initiate a new transient. There are no credible unanalyzed transients which will occur as a result of a dropped rod. The removal of this trip will reduce the potential for spurious or unnecessary trips which may occur as a result of maintenance or the drop of a low-worth rod. There are no other hardware modifications or procedure changes that will be made as a result of this deletion which could create the possibility of a new accident.

No margin of safety will be reduced by this change. As noted above, if a dropped rod necessitates a trip, the trip function will be accomplished as a result of low pressurizer pressure. For those dropped rods for which no trip is necessary, the removal of this trip will provide protection against an unnecessary transient.

LOW STEAM LINE SETPOINT PRESSURE CHANGE

Changing the Low Steam Line Pressure setpoint and removal of dynamic compensation will not increase the probability or consequences of any previously-reviewed accident. The higher steam line pressure setpoint is consistent with all licensing basis safety analyses. This change, in conjunction with the removal of the dynamic compensation of the steam pressure signal, is intended to reduce or eliminate spurious Engineered Safeguards Features (ESF) actuations which are caused by minor (but rapid) pressure decreases in the secondary system.

The proposed amendment will not result in a new accident not previously reviewed. A change in steam line pressure is a response to an existing transient condition, rather than a precursor or initiating event. A change in the steam line pressure setpoint is also not a precursor or initiating event.

The proposed amendment will not result in a significant decrease in a margin of safety. The reanalysis of the steam line break accident which was performed shows that all imposed Condition II acceptance criteria are met.

Based on the above, it is concluded that no significant hazards exist.

FEEDWATER AND MAIN STEAM LINE ISOLATION VALVE STROKE TIME

The proposed changes to the valve stroke times in Table 3.3-5 and Table 3.6-2a will not significantly increase the probability or consequences of any previously evaluated accident. The effects of the delays in isolation times on the various transients affected have been analyzed and found to be acceptable. Since these valves do not receive a containment isolation signal, and no credit is taken for operation of these valves in the dose analysis for a containment isolation function, a maximum stroke time does not apply for containment isolation.

The proposed changes will not significantly increase the possibility of a new accident not previously evaluated. Feedwater and main steam isolation are responses to ongoing transients, rather than initiators or precursors of transients. No equipment or component reconfiguration will occur as a result of this change.

The proposed changes will not significantly decrease any margin of safety. As noted above, the effects of the longer isolation times have been evaluated and found to be acceptable.

Based on the above, it is concluded that no significant hazards exist.

INCREASE IN PRESSURIZER CODE SAFETY VALVE SETPOINT TOLERANCES

The proposed amendment will not result in a significant increase in the probability or consequences of any previously analyzed accident. The valve lift setting is challenged only after a transient has been initiated and is not a contributor to the probability of any transient or accident. The transients which involve pressure increases which would potentially challenge the safety valves have been analyzed to determine the consequences of delayed or premature valve actuation at the extremes of the new setpoint tolerances. These analyses show that all applicable acceptance criteria are met using the wider tolerances.

The proposed amendment will not result in the creation of any new accident not previously evaluated. As noted above, the setpoint tolerance only affects the time at which the safety valve opens following or during a transient, and is not a contributor to the probability of an accident.

The proposed amendment will not result in a significant decrease in a margin of safety. The limiting transient in each accident category has been analyzed to determine the effect of the change in lift setpoint tolerance on the transient. In each case, the results of the analyses met all acceptance criteria.

Based on the above, it is concluded that no significant hazards exist.

CONTAINMENT ISOLATION VALVES

The proposed changes to the valve stroke times in Table[s] 3.6-2a and 3.6[-]2b will not significantly increase the probability or consequences of any previously evaluated accident. The effects of the delays in isolation times on the various transients affected have been analyzed and found to be acceptable. Since these valves do not receive a containment isolation signal, and no credit is taken for operation of these valves in the dose analysis for a containment isolation function, a maximum stroke time does not apply for containment isolation.

The proposed changes will not significantly increase the possibility of a new accident not previously evaluated. Feedwater and main steam isolation are responses to ongoing transients, rather than initiators or precursors of transients. No equipment or component reconfiguration will occur as a result of this change.

The proposed changes will not significantly decrease any margin of safety. The isolation times which are applicable to these valves are specified in Table 3.3-5, Engineered Safety Features Response Times. The effects of the isolation of these valves was evaluated based on their ESF function, not a containment isolation function, and determined to be acceptable, therefore there is no significant decrease in the margin of safety.

BORON DILUTION MITIGATION SYSTEM

TS 3.3.3.11.a.2 is changed to reduce the allowable Reactor Makeup Water Pump flow in Mode 5 from 75 gpm to 70 gpm. In the event that the Boron Dilution Mitigation System (BDMS) is inoperable the Reactor Makeup Water Pump flowrates are limited to ensure that operator action times required to terminate a dilution event can be met. The limits on reactor makeup water pump flowrates when the BDMS is inoperable are verified each cycle to ensure that the safety analysis assumptions for these parameters remain valid. When the calculated Reactor Makeup Water Pump flowrate is found to be less than the existing flowrate limits, the flowrate limit must be reduced so that the operator action time acceptance criteria of Standard Review Plan 15.4.6 can be met.

Reducing the allowable Reactor Makeup Water Pump flow in Mode 5 does not involve a significant increase in the probability or consequences of an accident previously evaluated. The current TS flowrate does not allow enough time for the operator to terminate an uncontrolled dilution event when required operator response times are assumed. The lower flowrate allows needed operator response times and is therefore more conservative.

Reducing the allowable Reactor Makeup Water Pump flow in Mode 5 does not change the way that any plant equipment is operated or maintained, therefore it does not create the possibility of a new or different accident.

Reducing the Allowable Reactor Makeup Water Pump Flow in Mode 5 will not involve a significant reduction in the margin of safety. This flowrate is more conservative, and ensures that safety analysis assumptions regarding operator actions times in response to the termination of an uncontrolled dilution event can be met.

CORE OPERATING LIMITS REPORT

The proposed change to TS 6.9.1.9 adds approved topical DPC-NE-1004A to the list of analytical methods used to determine core operating limits. This change is administrative, adding a topical report which has been approved for use on Catawba to the list of analytical methods used to determine core operating limits. Since this change is administrative it has been determined that no significant hazards are involved.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied for those changes. In addition, the NRC staff finds that administrative changes including deleting pages no longer applicable to Unit 2, and the renumbering and redesignation of remaining pages in certain sections as applicable to both units, will not involve a significant increase in the probability or consequences of an accident previously evaluated. These changes will not create the possibility of a new or different kind of accident from any accident previously evaluated for the use of BWFC fuel and the revised analysis methodology in the Catawba units. These changes will not involve a significant reduction in a margin of safety since they reflect the usage of fuel and analysis methodology that have been previously approved for the Catawba units. Therefore, on the basis of the licensee's and the staff's discussions above, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. The Commission will not normally make a final determination unless it receives a request for a hearing.

Written comments may be submitted by mail to the Rules Review & Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this FEDERAL REGISTER notice. Written comments may also be delivered to Room P-223, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555. The filing of requests for hearing and petitions for leave to intervene is discussed below.

By March 26, 1993 , the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room located at the York County Library, 138 East Black Street, Rock Hill, South Carolina 29730. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the FEDERAL REGISTER a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to David B. Matthews: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this FEDERAL REGISTER notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendments dated December 15, 1992, as supplemented February 5, 1993, which are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room located at York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

Dated at Rockville, Maryland, this 18th day of February 1993.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert E Martin

Robert E. Martin, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation