July 14, 1992

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

Dear Mr. Tuckman:

SUBJECT: CATAWBA UNITS 1 AND 2 - UNIT 1 CYCLE 7 RELOAD (TAC NOS. M83173 AND M83174)

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

This notice relates to your application dated April 13, 1992, as supplemented July 8, 1992, to change the Technical Specifications for the Unit 1 Cycle 7 reload.

Sincerely,

L.A.WIENS for/R.MARTIN

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

Enclosure:
As stated

cc w/enclosure:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

July 14, 1992

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and 50-414

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Sincerely,

[signature] for

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

Enclosure:
As stated

cc w/enclosure:
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UNITED STATES NUCLEAR REGULATORY COMMISSION

DUKE POWER COMPANY

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

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 Duke Power Company (the licensee) for operation of the Catawba Nuclear Station, Units 1 and 2 located in York County, South Carolina.

The proposed amendment would revise the Technical Specifications for Unit 1 Cycle 7 reload. Cycle 7 for Catawba Unit 1, scheduled to begin in September 1992, is the second Catawba Cycle for which the reload fuel is supplied by B&W Fuel Company (BWFC). The incoming Batch 9 fuel assemblies are designated as Mark-BW. To support implementation of Mark-BW fuel in the McGuire and Catawba nuclear stations, Duke Power Company (DPC) has developed new methods and models to analyze the plants during normal and off-normal operation. The thermal-hydraulic analytical models are documented in topical report DPC-NE-3000P for non-LOCA transients and BAW-10174 for LOCA. Portions of the analytical methodology are documented in topical report DPC-NE-3001P and DPC-NE-2004PA. The remaining Final Safety Analysis Report (FSAR) Chapter 15 non-LOCA system transient analysis methodology is documented in DPC-NE-3002. The FSAR Chapter 15 LOCA system transient analysis methodology is documented in BAW-10174. The NRC staff has issued safety evaluations on these topical reports.

The licensee states that all of the accidents analyzed in the FSAR have been reviewed for Cycle 7 operation, and that many of the FSAR Chapter 15 system thermal-hydraulic accident analyses sensitive to reload core physics parameters have been reanalyzed using Duke Power methodology. Several bounding transients were analyzed in detail to demonstrate the capability of DPC calculational techniques. The results of these analyses were reported in DPC-NE-3001P. For the other reanalyzed transients, the approved methodology is documented in DPC-NE-3002. The Technical Specifications (TS) that the licensee proposes to be changed are as follows:

Specification	Description of Change
2.1.1, 2.2.1	Decreased F_{deltaH} for Mark-BW fuel Removed power range neutron flux negative rate reactor trip Removed Total Allowance, Z value, and Sensor Error terms
3.1.3.1	Included all accident analyses that would require reevaluation in the event that one full length RCCA is inoperable
3/4.2.2, 3/4.2.3	Changed F_Q and F_{deltaH} methodology to reflect Duke nomenclature Quantified surveillance requirements
3/4.2.5	Corrected action item requirement
3/4.3.3.1	Removed power range neutron flux negative rate reactor trip Removed items associated with RTD Bypass System
3/4.3.3.2	Increased low steam line pressure setpoint Increased feedwater isolation response time

Specification	Description of Change
3/4.3.3.2 (cont.)	Increased steam line isolation response time Removed Total Allowance, Z value, and Sensor Error terms Removed steam line pressure dynamic compensation
3/4.4.1.2	Changed reactor coolant loop operation requirement
3/4.4.2.1, 3/4.4.2.2	Increased pressurizer safety valve lift setpoint tolerance
3/4.5.1.c	Changed required cold leg accumulator boron concentration
3/4.5.2	Changed ECCS pump surveillance requirements
3/4.6.2	Reduced allowable primary to secondary leakage rate
3/4.6.3	Changed feedwater isolation valve, main steam isolation valve, and main steam isolation bypass valve stroke time from 5 seconds to Not Applicable
3/4.7.1.4	Increased main steam line isolation valve stroke time
6.9.1.9	Reflected change to DPC core operating limit methodology

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. The following was included in the licensee's analysis.

* * * *

POWER DISTRIBUTION AND SAFETY LIMITS

* * * *

The Catawba Unit 1, Cycle 7 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 7, the core will contain 72 fresh fuel assemblies, 72 burned fuel assemblies supplied by B&W and 49 Westinghouse supplied Optimized Fuel Assemblies (OFA).

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.

* * * *

Duke Power Company's Topical Reports DPC-NE-3000, DPC-NE-3001, and DPC-NE-2004 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 7 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 7 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.

An administrative change is being made to TS Tables 2.2-1 (Reactor Trip System Instrumentation Trip Setpoints), and Table 3.3-4 (Engineered Safety Features Actuation System Instrumentation Trip Setpoints). Since these tables contain values that are not identical for each unit, a separate table will be provided for each unit. The pages will be labeled "Unit 1" or "Unit 2", and there will be an "A" in the page number for Unit 1 and a "B" in the page number for Unit 2. The TS Tables will be copied on white paper for Unit 1 and on yellow paper for Unit 2 to further distinguish applicability. Table 3.3-4 will also have references to the RTD bypass system deleted, since the RTD bypass system has been removed, and they no longer apply. These changes are administrative in nature, and are being made only to clarify the TS. Since they involve no change in requirements, they involve no significant hazards.

REMOVAL OF TOTAL ALLOWANCE Z AND SENSOR ERROR ...

The removal of the Total Allowance, Sensor error, and Z columns from Tables 2.2-1 and 3.3-4, along with the deletion of TS 2.2.1.b.1, 3.3.2.b.1, and equation 2.2-1, which provide for the use of these values, do not involve any significant hazards consideration. These specifications provide the option of declaring instrumentation operable when the setpoint is less conservative than the allowable value. This is done through the use of equation 2.2-1. With the deletion of Specifications 2.2.1.b.1, 3.3.2.b.1, equation 2.2-1, and the Total Allowance, Sensor Error, and Z columns from Tables 2.2-1 and 3.3-4 the channel must be declared inoperable with the setpoint less conservative than the Allowable Value. This change is more conservative than the current requirements, and therefore involves no significant hazards.

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.

REDUCE ALLOWABLE PRIMARY TO SECONDARY LEAKAGE

The allowable primary to secondary leakage has been reduced to limit the offsite radiological dose consequences due to the reanalysis of the locked rotor, rod ejection, and single uncontrolled rod withdrawal FSAR Chapter 15 events. The new limits are more conservative than the current TS requirements. Lowering the allowable primary to secondary leakage will not increase the probability of a previously evaluated accident, it will ensure that the dose consequences of an accident are within allowable limits. The possibility of a new or different accident from any previously evaluated is not created because there will be no physical changes to the plant operating procedures, other than to more conservatively limit leakage. There will not be a significant reduction in the margin of safety due to the fact that the allowable leakage is more conservative.

Based on the above, it is concluded that no significant hazards are associated with this change.

INCREASE IN OPERABLE RCS LOOPS IN MODE 3 AND INCREASE COLD LEG ACCUMULATOR REQUIRED BORON CONCENTRATION

These amendments will not involve any significant hazards consideration. The proposed changes will result in the parameter or operating condition involved becoming more conservative than the current TS requirement. The NRC's own guidance, published in the Federal Register (48CFR 14870), states that an amendment which results in conditions becoming more restrictive is not likely to result in significant hazards consideration as defined by 10 CFR 50.92. Therefore, it may be concluded, with no further analysis, that these amendments will not involve a significant hazards consideration.

ECCS PUMP PERFORMANCE REQUIREMENTS

The proposed amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated because the Loss-of-Coolant-Accident (LOCA) analysis, to which the ECCS flowrates are input assumptions, is unchanged and, therefore, continues to meet applicable acceptance criteria.

The proposed amendments will not result in a significant increase in the possibility of a new accident because the new values represent a change in required pump performance. The new values represent no change in the assumptions made in the LOCA analysis, or any a physical change in the plant. Enough margin exists between the flow used in the LOCA analysis and the new required pump flows that a reanalysis was not necessary.

[T]he proposed changes will not result in a significant decrease in a margin of safety, because pump performance at the new values is sufficient to meet all acceptance criteria in both the current FSAR analysis and any analysis associated with Catawba 1 Cycle 7.

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.

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 Tables 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 recieve 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.

REVISE LIST OF ACCIDENTS REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE RCCA

The proposed change to Table 3.3-1 will not change the probability or consequences of any accident or reduce any safety margin, because the table simply lists accident analyses which must be reevaluated in the event of an inoperable rod cluster control assembly (RCCA). The activities involved are analytical only, and do not introduce any operational considerations. Revision of the table to more accurately define the affected analyses is an administrative effort related to activities (analyses) which are conducted offsite after the fact of a postulated inoperable RCCA.

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

* * * *

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. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The licensee has also proposed changes to TS 6.9.1.9 to update the listing of previously approved topical reports which describe the analytical methods used to determine the core operating limits. This updating is an administrative change that provides consistency between the list and the changes made as discussed above in the prior sections of the TS. Accordingly, the updating of this list to reflect the titles of the reports describing the underlying methodology for the changes discussed above does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the the possibility of a new or different kind of accident from any previously evaluated, and does not involve a significant reduction in a margin of safety. On this basis the staff proposes to find that this change does not involve a significant hazards consideration. The Commission is seeking public comments on this proposed determination. Any comments received within thirty (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 and Directives Review 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 August 20, 1992 , 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 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, 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 fifteen (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 fifteen (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 ten (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) 325-6000 (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 amendment dated April 13, 1992, as supplemented July 8, 1992, which is 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 **14th** day of **July 1992.**

FOR THE NUCLEAR REGULATORY COMMISSION

Leonard A. Wiens, Acting Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation