AUG 2 5 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 NUCLEAR STATION, UNIT 1 - STEAM GENERATOR INTERIM PLUGGING CRITERIA (TAC NOS. M84221 AND M84222)

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

This notice relates to your August 24, 1992, application to change Technical Specification (TS) Sections 3/4.4.5 Steam Generators, and 3/4.4.6 Reactor Coolant System Leakage along with their associated Bases to revise the repair criteria for Unit 1 Steam Generators for Catawba Unit 1 Cycle 7 operation. The proposed change would allow the use of an interim tube plugging criteria, which will utilize a bobbin voltage-based plugging criteria.

Sincerely,

ORIGINAL SIGNED BY:

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Notice

cc w/enclosure:
See next page

DISTRIBUTION

Docket File	GLainas	OC/LFMB
NRC/Local PDRs	OPA	JStang
PDII-3 Reading	ACRS (10)	
Catawba Reading	SVarga	
DMatthews	OGC	
RMartin	DHagan	
LBerry	EMerschoff, RII	

OFFICE:	PDII-3/LA	PDII-3/PM	PDII-3/D
NAME:	L.BERRY	for R.MARTIN	D.MATTHEWS
DATE:	8/25/92	8/25/92	8/25/92

OFFICIAL RECORD COPY
FILE NAME: A:CATSG.LTR

9209100145 920825
PDR ADOCK 05000113
P PDR

OP-1

DF01 1/1

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

August 25, 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 NUCLEAR STATION, UNIT 1 - STEAM GENERATOR INTERIM PLUGGING CRITERIA (TAC NOS. M84221 AND M84222)

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

This notice relates to your August 24, 1992, application to change Technical Specification (TS) Sections 3/4.4.5 Steam Generators, and 3/4.4.6 Reactor Coolant System Leakage along with their associated Bases to revise the repair criteria for Unit 1 Steam Generators for Catawba Unit 1 Cycle 7 operation. The proposed change would allow the use of an interim tube plugging criteria, which will utilize a bobbin voltage-based plugging criteria.

Sincerely,

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Notice

cc w/enclosure:
See next page

Mr. M. S. Tuckman
Duke Power Company

Catawba Nuclear Station

cc:
Mr. R. C. Futrell
Regulatory Compliance Manager
Duke Power Company
4800 Concord Road
York, South Carolina 29745

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

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

North Carolina MPA-1
Suite 600
P. O. Box 29513
Raleigh, North Carolina 27626-0513

Mr. Frank Modrak
Project Manager, Mid-South Area
ESSD Projects
Westinghouse Electric Corporation
MNC West Tower - Bay 241
P. O. Box 355
Pittsburgh, Pennsylvania 15230

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

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

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

Mr. Alan R. Herdt, Chief
Project Branch #3
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW. Suite 2900
Atlanta, Georgia 30323

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

Senior Resident Inspector
Route 2, Box 179 N
York, South Carolina 29745

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

Mr. Heyward G. Shealy, Chief
Bureau of Radiological Health
South Carolina Department of Health and Environmental Control
2600 Bull Street
Columbia, South Carolina 27602

Mr. R. L. Gill, Jr.
Licensing
Duke Power Company
P. O. Box 1007
Charlotte, North Carolina 28201-1007

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

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

UNITED STATES NUCLEAR REGULATORY COMMISSION

DUKE POWER COMPANY, ET AL.

DOCKET NOS. 50-413 AND 50-414

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

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License Nos. NPF-35 issued to the Duke Power Company, et al. (the licensee), for operation of the Catawba Nuclear Station, Unit 1 located in York County, South Carolina.

The proposed amendment would change Technical Specification (TS) Sections 3/4.4.5 Steam Generators, and 3/4.4.6 Reactor Coolant System Leakage along with their associated Bases to revise the repair criteria for Unit 1 Steam Generators for Catawba Unit 1 Cycle 7 operation. The proposed change would allow the use of an interim tube plugging criteria, which will utilize a bobbin voltage-based plugging criteria.

The licensee is requesting that this amendment be processed on an exigent or, if necessary, emergency basis pursuant to 10 CFR 50.91(a)(5) or (6). The licensee states that during the Unit 1 end of cycle 6 refueling outage, which is currently underway, Catawba began its inspection of Unit 1 steam generators. The following was provided by the licensee in support of their request:

> Bobbin coil inspections of the steam generator tubes were completed by August 8, 1992. [Approximately 7000 indications were found which affected approximately 4500 tubes.] When an indication is found

using the bobbin coil technique, the Motorized Rotating Pancake Coil (MRPC) is used to confirm the existence of the indication. Use of the MRPC on a sample population of Catawba Unit 1 tubes confirmed the presence of indications in approximately 23% of those tubes sampled. This effort was completed, and the data was available, on August 10, 1992. Using this confirmation [sic] data and the current criteria required by the Catawba Technical Specifications, Catawba has projected that approximately 1020 tubes would require repair.

* * * *

With this data available, and after balancing these considerations, Catawba management decided on August 11, 1992, to pursue the possibility of amending Unit 1's Tech Specs to permit the use of interim plugging criteria. On August 11, 1992, Duke requested Westinghouse to begin its analyses to support such a change. That same day, Duke also contacted the NRC Staff to inform them of the results of the steam generator inspection and analyses.

During the August 11, 1992, conversation, Duke and the Staff discussed a preliminary schedule for development and submittal of the proposed Tech Spec change and its justification. A date of August 14, 1992, was tentatively set for submittal of the application, to include the proposed Tech Spec pages and a No Significant Hazards analysis. Because of the complexity of the analyses involved, Duke and Westinghouse were unable to meet this schedule and on August 14, 1992, Duke so informed the Staff. Late the afternoon of August 19, 1992, Duke received draft analyses and submittals for review from Westinghouse. Since that time Duke and Westinghouse have been engaged in an iterative process of reviewing and developing the pertinent documents and analyses to assure, among other things, that the assumptions made by Westinghouse in its analyses are consistent with the accident and dose analyses used by Duke in the licensing of Unit 1. During this entire process, Catawba has been in a daily telephone contact with NRC Staff to keep the Staff informed on the progress of this Tech Spec submittal.

In sum, grant of the proposed amendments to the Unit 1 Tech Specs to allow implementation of the Interim Tube Plugging Criteria will, by decreasing the inspection and repair requirements under the existing Tech Specs:

- Save about 100 days in unplanned refueling outage time

- Reduce projected personnel exposures by approximately 45 person-rem

- Save approximately 8 million dollars, and

- Maintain a larger Reactor Coolant flow margin

Therefore, for the reasons set out above, Duke requests that this amendment be processed on an exigent or, if necessary, an emergency basis as provided in 10 CFR 50.91(a)(5) or (6). The steam generator tube inspections and repairs required during the current outage under existing Tech Specs could not have been projected by Duke based on the plant-specific and industry-wide data available prior to the outage. When, during the outage, actual inspections showed that the number of needed inspections and repairs could significantly exceed its projections, Duke took immediate action to develop the Interim Tube Plugging Criteria for Unit 1. The proposed amendment is necessary to meet the schedule for return to operation of Unit 1. This requested Tech Spec amendment has been pursued in a timely manner and in full consultation with the NRC Staff. The need for exigent or, if necessary, emergency processing of this Tech Spec amendment was not because of dilatory behavior on the part of Duke Power Company.

The licensee transmitted their application to the NRC on August 25, 1992. Catawba Unit 1 is currently scheduled to enter Mode 4 on or about September 12, 1992, and this amendment will be necessary to declare the Steam Generators operable at that time. Consequently, it will be necessary to issue this amendment in order not to delay startup of the unit. This schedule does not provide the requisite time for the publication of the appropriate Notice in the Federal Register for the 30-day period pursuant to 10 CFR 50.91(a)(2)(ii). The staff has reviewed the schedular information and the actions undertaken by the licensee and has decided to process the amendment on an exigent basis because a failure to do so would result in a delay in the startup of the unit past the currently scheduled date. Based on the information provided, it appears that the licensee's actions have reflected their best efforts to make a timely application for the needed changes to the TSs.

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:

> In accordance with the three factor test of 10 CFR 50.92(c), implementation of the proposed license amendment is analyzed using the following standards and found not to: 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 margin of safety.
>
> Conformance of the proposed amendment to the standards for a determination of no significant hazard as defined in 10 CFR 50.92 (three factor test) is shown in the following:
>
> 1) Operation of Catawba Unit 1 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.
>
> Testing of model boiler specimens for free span tubing (no tube support plate restraint) at room temperature conditions show burst pressures in excess of 5475 psi for indications of outer diameter stress corrosion cracking with voltage measurements as high as 11 volts (Reference 1). Burst testing performed on pulled tubes from Catawba Unit 1 with up to a 1.5 volt indications show measured burst pressures in excess of 4800 psi

at room temperature. Correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature, tube burst capability significantly exceeds the R.G. 1.121 criterion requiring the maintenance of a margin of 3 times normal operating pressure differential on tube burst. The 3 times normal operating pressure differential for the Catawba Unit 1 steam generators corresponds to 3750 psi. Based on the existing data base, this criterion is satisfied with 3/4" diameter tubing with bobbin coil indications with signal amplitudes less than 4.1 volts, regardless of the indicated depth measurement. This structural limit is based on a lower 95% confidence level limit of the data. A 1.0 volt plugging criterion compares favorably with the structural limit considering the calculated growth rates for ODSCC within the Catawba Unit 1 steam generators. Considering a voltage increase of 0.58 volts, and adding 20% NDE uncertainty of 0.2 volts (90% Cumulative Probability) to the interim plugging criterion of 1.0 volts results in an EOC voltage of 1.78 volts. The growth rate used to determine the projected EOC voltage is based on the review of growth rates for 541 TSP intersections. These indications were selected by Duke Power Company based on their largest amplitudes from the original analyses. The 541 indications were made up of 90, 117, 197, and 137 from steam generators A, B, C and D, respectively. This end of cycle voltage compares favorably with the Structural Limit 4.1 volt. The corresponding safety margin to the tube structural limit at end of cycle 7 upon implementation of the 1.0 volt steam generator tube interim plugging limit is 2.3 volts. The necessary plugging limit to meet tube structural limits is 2.5 volts.

Only three indications of ODSCC have been reported to have operating leakage - all three have been in European plants. No field leakage has been reported at other plants from tubes with indications with a voltage level of under 6.2 volts (from 3/4" tubing). Relative to the expected leakage during accident condition loadings, the accidents that are affected by primary to secondary leakage and steam release to the environment are: Feedwater System Malfunction, Loss of External Electrical Load and/or Turbine Trip, Loss of All AC Power to Station Auxiliaries, Uncontrolled Single Rod Withdrawal at Power, Major Secondary System Pipe Failure, Steam Generator Tube Rupture, Reactor Coolant Pump Locked Rotor, and Rupture of a Control Rod Drive Mechanism Housing. In support of implementation of the interim plugging criterion, it has been determined that the distribution of cracking indications at the tube support plate intersections at the end of cycle 7 are projected to be such that primary to secondary leakage would result in site boundary doses within a small fraction of the 10 CFR 100 guidelines.

Monte Carlo analyses methods are used to calculate the potential SLB leakage at the EOC-7 at Catawba Unit 1. The Monte Carlo analyses methods utilize the distributions for indications left inservice, NDE uncertainties, voltage growth and SLB leak rate. The methods account for the tails of the distribution and yield eddy current voltages with an associated probability of occurrence and the cumulative probability of EOC voltages. The SLB leak rates applied to the Monte Carlo voltage distribution are 0.0 gpm for volts less than or equal to 1.8 volts, 1 liter/hr for 1.8 to 3.5 volts, and 10 liter/hr for greater than 3.5 volts. Applying these leak rates to the projected EOC voltage distribution leads to a projected SLB leak rate of 0.54 gpm for steam generator D, the most limiting steam generator (3492 TSP elevation indications). The 0.54 gpm SLB leak rate compares favorably with the accident analyses assumptions of 1.0 gpm in the affected steam generator identified in Table 15.3 of the Catawba Unit 1 Safety Evaluation Report. The projection indicates a maximum EPC-7 of 3.1 volts (90% cumulative probability). The analyses yields a negligible likelihood of a tube exceeding the 3.5 volt threshold for a 10 liter/hr SLB leak rate.

Upon application of the interim plugging criterion, only a negligible increase in leakage above normal operating leakage would be expected during plant transients, other than steam line break, which have lower peak differential pressures.

Therefore, as steam generator tube burst capability and leaktightness during Cycle 7 operation following implementation of the proposed 1.0 volt interim plugging criterion remains consistent with the current licensing basis, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated with the Catawba Unit 1 FSAR.

2) The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementation of the proposed interim tube support plate elevation steam generator tube plugging criterion does not introduce any significant changes to the plant design basis. Use of the criterion does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations; no ODSCC is occurring outside the thickness of the tube support plates. A tube rupture event would not be expected in a steam generator in which the plugging criterion has been applied (during all plant conditions).

Upon application of the interim plugging criterion, no primary to secondary leakage during normal operating is anticipated during all plant conditions due to degradation at the tube support plate elevations in the Catawba Unit 1 steam generators. However, additional conservatism is built into the operating leakage limit with regard to protection against the maximum permissible single crack length which may be achieved during Cycle 7 operation due to the potential occurrence of through wall cracks at locations other than the tube support plate intersections.

Specifically, Duke Power Company will implement a maximum leakage rate limit of 150 gpd (0.1 gpm) per steam generator to help preclude the potential for excessive leakage during all plant conditions. The currently proposed Cycle 7 Reload Technical Specification limits on primary to secondary leakage at operating conditions is a maximum of 0.5 gpm (720 gpd) for all steam generators, or, a maximum of 200 gpd for any one steam generator. The R.G. 1.121 criterion for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture. The 150 gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. R.G. 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of 3 against bursting at normal operating pressure differential. A voltage amplitude of 4.1 volts for typical ODSCC corresponds to meeting this tube burst requirement at a lower 95% uncertainty limit on the burst correlation. Alternate crack morphologies can correspond to 4.1 volts so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, typical burst pressure versus through-wall crack length correlations are used below to define the "longest permissible crack" for evaluating operating leakage limits.

The single through-wall crack lengths that result in tube burst at 3 times normal operating pressure differential and SLB conditions are 0.48 inch and 0.76 inch, respectively. Nominal leakage for these crack lengths would range from about 0.10 gpm to 3 gpm, respectively, while lower 95% confidence level leak rates would range from about 0.015 gpm to 0.4 gpm, respectively. A leak rate of 150 gpd will provide for detection of 0.40 inch long cracks at nominal leak rates and 0.60 inch long cracks at the lower 95% confidence level leak rates.

Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for SLB conditions at leak rates less than a lower 95% confidence level and for three times normal operating pressure differential at less than nominal leak rates.

Application of the 1.0 volt interim steam generator tube plugging criterion at Catawba Unit 1 is not expected to result in tube burst during all plant conditions during Cycle 7 operation. Tube burst margins are expected to meet R.G. 1.121 acceptance criteria. The limiting consequence of the application of the interim plugging criterion is a potential for primary to secondary leakage of approximately 0.54 gpm. This amount of leakage does not result in unacceptable radiological consequences. No unacceptable leakage is anticipated at normal operating or RCP locked rotor conditions. Therefore, as the existing tube integrity criteria and accident analyses assumptions and results continue to be met, the proposed license amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

3) The proposed license amendment does not involve a significant reduction in margin of safety.

Based on the analysis which shows the new leakage values proposed and the leakage characteristics expected during accidents creating high differential pressures across the steam generator tubes (main steam line break) new dose analyses were run to determine offsite dose consequences. A new analysis of the Main Steam Line Break accident using pre-existing leakage's of 0.1 gpm per steam generator and leakage growth of 1.1 gpm in the faulted generator determined that the EAB and Low Population Zone doses remain well within 10% of the allowed 10 CFR100 values of 25 Rem whole body and 300 Rem thyroid. The most restrictive dose analysis is the Reactor Coolant Pump Locked Rotor accident which requires that total steam generator leakage remains less than 0.7 gpm. This is a new analysis which has been submitted to support Unit 1 Cycle 7. This accident does not create excessive differential pressure conditions across the steam generator tubes and by limiting the initial allowed primary to secondary leakage to 0.4 gpm total, 10% of 10 CFR100 dose limits are again not exceeded. Reruns of the above accident dose analyses show that there is no significant increase in dose consequences.

The use of the voltage based bobbin probe interim tube support plate elevation plugging criterion at Catawba Unit 1 is demonstrated to maintain steam generator tube integrity commensurate with the criteria of Regulatory Guide 1.121. R.G. 1.21 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criterion, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The end of cycle distribution of crack indications at the tube support plate elevations is calculated to result in minimal primary to secondary leakage during all plant conditions and radiological consequences are not adversely impacted.

In addressing the combined effects of LOCA + SSE on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

Analyses results show that for the Catawba Unit 1 steam generators several tubes near wedge locations may significantly deform or collapse and secondary to primary inleakage may result. These tubes have been precluded from application of interim plugging criterion (Reference 3). For all other steam generator tubes, the possibility of secondary to primary leakage in the event of a LOCA + SSE event is not significant. In actuality, the amount of secondary to primary leakage in the event of a LOCA + SSE is expected to be less than that associated with the application of this criterion, i.e., 150 gpd per steam generator. Secondary to primary inleakage would be less than primary to secondary leakage for the same pressure differential since the cracks would tend to close under a secondary to primary pressure differential. Additionally, the presence of the tube support plate is expected to reduce the amount of in-leakage.

Addressing R.G. 1.83 considerations, implementation of the bobbin probe voltage based interim tube plugging criterion of 1.0 volt is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100% eddy current inspection sample size at the tube support plate elevations, and rotating pancake coil inspection requirements for the larger indications left inservice to characterize the principal degradation as ODSCC.

As noted previously, implementation of the tube support plate elevation plugging criterion will decrease the number of tubes which must be repaired or taken out of service by plugging. The installation of steam generator tube plugs or sleeves reduces the RCS flow margin. Thus, implementation of the alternate plugging criterion will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any BASES of the plant Technical Specifications.

CONCLUSION

Based on the preceding analysis, it is concluded that using the TSP elevation bobbin coil probe voltage-based interim steam generator tube plugging criterion for removing tubes from service at Catawba Unit 1 is acceptable and the proposed license amendment does not involve a Significant Hazards Consideration Finding as defined in 10 CFR 50.92.

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 Commission is seeking public comments on this proposed determination. Any comments received within fifteen (15) 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 September 28, 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 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 a 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 the amendment is issued before the expiration of 30-days, the Commission will make a final determination on the issue of no significant hazards consideration. If a hearing is requested, 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 15-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 15-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. 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 August 24, 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 the York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

Dated at Rockville, Maryland, this 25th day of August 1992.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation