February 23, 1993

Docket No. 50-395

Mr. John L. Skolds
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065

Dear Mr. Skolds:

SUBJECT: NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR HEARING - VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 (TAC NO. M85792)

Enclosed is a copy of an individual notice for your information. This notice relates to your application dated February 12, 1993, which would modify Technical Specifications 3/4.4.5, Steam Generators, 3/4.4.6, Reactor Coolant System Leakage, and the associated bases to allow the use of interim plugging criteria at Virgil C. Summer Station, Unit No. 1.

This notice has been forwarded to the Office of the Federal Register for publication.

Sincerely,

ORIGINAL SIGNED BY:

George F. Wunder, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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Notice

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UNITED STATES NUCLEAR REGULATORY COMMISSION

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

DOCKET NO. 50-395

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 No. NPF-12 issued to South Carolina Electric & Gas Company (the licensee) for operation of the Virgil C. Summer Nuclear Station, Unit No. 1 (V. C. Summer), located in Fairfield County, South Carolina.

The proposed amendment would allow the use of interim plugging criteria (IPC) at V. C. Summer; specifically it would modify Technical Specifications 3/4.4.5, Steam Generators, 3/4.4.6, Reactor Coolant System Leakage, and the associated bases which provide tube inspection requirements and acceptance criteria to determine the level of degradation for which a tube experiencing outside diameter stress corrosion cracking (ODSCC) at the tube support plate elevations may remain in service in the V. C. Summer steam generators. For Cycle 8 operation of V. C. Summer, an interim tube support plate elevation plugging criteria that uses a voltage based plugging limit is proposed; specifically, flaw indications with a bobbin coil voltage less than or equal to 1.0 volt can remain in service without further action. For flaw indications in excess of 1.0 volt but less than 2.2 volts, the tube can remain in service provided a rotating pancake coil (RPC) inspection of the indication does not detect a defect. Indications involving modes of tube degradation other than ODSCC will be assessed for operability based on the current plugging criteria.

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:

1. Operation of V. C. Summer 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 (WCAP-13522). Burst testing performed on pulled tubes from similar plants with 3/4 inch tubing shows a measured burst pressure of approximately 5400 psi at room temperature for an indication at 3.5 volts. 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 RG [Regulatory Guide] 1.121 criterion requiring the maintenance of a margin of 3 times normal operating pressure differential (3 [delta] P) on tube burst. The 3 [delta] P for the V. C. Summer steam generators corresponds to 3996 psi. Bobbin voltages of tubes which burst near the 3 [delta] P limit on pressure differential ranged from 8.55 to 15.7 volts. Based on the existing data base, this criterion is satisfied with 3/4" diameter tubing with bobbin coil indications with signal amplitudes less than 3.7 volts, regardless of the indicated depth measurement. This structural limit is based on a least squares fit of the burst data at a lower 95% prediction interval, further reduced for operating temperature. Burst tube crack morphologies include tubes with single or only a few large axial cracks (more representative of Model Boilers), multiple axial crack networks which form larger macrocracks, and macrocrack networks which are formed predominantly of axial macrocracks with smaller, shallower circumferential cracks joining the axial macrocracks. This last morphology has been termed cellular corrosion. The burst data reduction conservatively bounds all data points from destructive examinations performed to date. No crack morphology detected has resulted in burst pressures for a particular voltage which is not representative of the entire data spread and not bounded by the voltage/burst correlation. Therefore, it is judged that the current voltage/burst correlation is representative of any observed or expected (by metalography) tube degradation morphology affecting the TSP [tube support plate] crevice area.

When considering the calculated growth rates for ODSCC within the V. C. Summer steam generators, a 1.0 volt plugging is shown to adequately protect the structural limit. Considering an average voltage increase from the previous cycles eddy current results of 0.29 volts, and adding a 15% NDE [non-destructive evaluation] uncertainty of 0.15 volts (15% of 1 volt) to the interim plugging criteria of 1.0 volts results in an EOC [end of cycle] voltage of 1.44 volts. This end of cycle voltage compares favorably with the structural limit of 3.7 volts. When 90% cumulative probability values for growth (0.70 volts) and 14% NDE uncertainty values (0.14 volt) are applied to the 1.0 volt plugging limit, the projected EOC voltage of 1.84 volts is obtained. From the voltage/burst correlation in WCAP-13522, the predicted burst pressure of a 1.84 volt indication is 4740 psi, 19% greater than the 3 [delta] P limit established by RG 1.121 of 3996 psi. The growth rate used to determine the projected EOC voltage is based on the review of growth rates for 87 TSP intersections. This includes all reported indications for which detectable eddy current signals were observed from both the 1990 and 1991 inspections.

Additionally, representing the bounding case, the maximum voltage increase from the previous cycle of 2.0 volts is applied to the 1.0 volt plugging limit, resulting in an EOC voltage of 3.0 volts, which is still below the EOC structural limit. Adding NDE uncertainty at 99% cumulative probability to the maximum growth results in a bounding EOC voltage of 3.25 volts, which is 0.45 volts below the RG 1.121 structural limit of 3.7 volts. It must further be noted that the RG 1.121 structural limit applies a factor of safety of 3 to the normal operating pressure differential. A more realistic comparison would be to compare this bounding EOC voltage of 3.25 volts to the voltage which would indicate a burst potential at SLB [steam line break]. Using the lower 99 % prediction interval burst correlation results in a burst pressure voltage for SLB conditions of approximately 9.3 volts.

The BOC [beginning of cycle] voltage limit of 2.2 volts is used to protect against EOC voltages exceeding the 3.7 volt, RG 1.121 structural limit. The 2.2 volt value represents an upper bound to assess the acceptability of bobbin coil indications, regardless of

RPC [rotating pancake coil] verification. It is conservatively assumed that BOC voltage levels of this magnitude and greater could result in an indication reaching the structural limit at EOC conditions. The 2.2 volt limit is derived by starting with the 3.7 volt structural limit and reducing it by the NDE uncertainty at 90% cumulative probability (20% uncertainty, .45 volts) and the average voltage growth (45% growth, 1.0 volt). This methodology is considered conservative since the 90% cumulative uncertainty value is found to be 14%, but conservatively assumed to be 20% for the analysis. Average growth values for all indications was 44%, while average growth from BOC indications above 0.75 volt was found to be 16%. The 1.0 volt growth allowance (45%) corresponds to 96% cumulative voltage growth. Use of a 1.0 volt interim plugging limit implies an inherent 1.2 volt margin to a plugging limit developed using RG 1.121 methodology.

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 7.7 volts (from 3/4" tubing). Relative to the expected leakage during accident condition loadings, the accident analyses affected by primary-to-secondary leakage and steam release to the environment as described in the V. C. Summer FSAR [Final Safety Analysis Report] will not be affected upon implementation of the IPC [interim plugging criteria].

A deterministic bobbin voltage-leakage potential correlation is applied to the Monte Carlo voltage predictions to calculate the potential SLB leakage at the EOC-8 at V. C. Summer. The leak rate data for the model boiler and pulled tube specimens were analyzed in order to establish an algebraic relationship that could be used to predict the probability of leakage as a function of bobbin voltage amplitude. A correlation between bobbin voltage and SLB leak rate has been established. The slope of the SLB leak rate correlation is affected by the selection of the regressor variable (voltage vs. leak rate or leak rate vs. voltage). The final correlation combines the portions of the two regression curves to obtain a conservative relationship over the entire voltage range. Based on the protected EOC voltage distribution established by the Monte Carlo analysis, a particular probability of leakage for each voltage bin (i.e., each voltage increment, usually kept to 0.1 volts for simplicity) is established. The values for SLB leak rate for each voltage bin are applied to the leakage probability for each bin. The total SLB leak rate is then determined by integrating the result of the entire EOC voltage distribution. Applying this methodology to the V. C. Summer projected EOC voltage distribution from the previous cycle results in a maximum leak rate of 0.02 gpm for loop B, the most limiting steam generator. The 0.02 gpm SLB leak rate compares favorably with the accident analyses assumptions of 1.0 gpm identified in Table 15.4-23 of the V. C. Summer FSAR. When the probability of leakage and leak rate at 95% confidence limits are applied to a 2.0 volt indication, it is determined that overall effect of each 2.15 volt

EOC indication would be to contribute approximately 0.0009 gpm per 2.15 volt indication during an SLB with a [delta] P of 2335 psia. The Monte Carlo analyses indicate a maximum EOC 7 projection of 2.97 volts (99.5% cumulative probability). This value is typical of the maximum voltages observed following application of the existing depth based criteria.

Upon application of the interim plugging criteria, only a negligible increase in leakage (if any increase at all is determined) above normal operating leakage would be expected during plant transients.

Therefore, as implementation of the proposed 1.0 volt interim plugging criteria during Cycle 8 does not adversely affect steam generator tube integrity and results in acceptable dose consequences, the proposed amendment does not result in any [significant] increase in the probability or consequences of an accident previously evaluated within the V. C. Summer 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 criteria does not introduce any significant changes to the plant design basis. Neither a single or multiple tube rupture event would be expected in a steam generator in which the plugging criteria has been applied (during all plant conditions). And use of the criteria does not provide a mechanism which could result in an accident that has not previously been evaluated.

Upon application of the interim plugging criteria, no excessive primary-to-secondary leakage is anticipated during all plant conditions due to degradation at the tube support plate elevations in the V. C. Summer steam generators. However, the RG 1.121 criterion of providing protection against the leakage from the maximum permissible single crack length which may be achieved during Cycle 8 operation must be met. The primary-to-secondary leakage limits proposed to be implemented with the interim plugging criteria is conservative with respect to the RG 1.121 criterion.

Concurrent with the implementation of the interim plugging criteria, SCE&G will implement a maximum operational leakage limit of 450 gpd for all steam generators, and a maximum of 150 gpd for any one steam generator. The leakage limits will help preclude the potential for excessive leakage during all plant conditions. The RG 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 crack length that provides a factor of safety of 3 against bursting at normal operating pressure differential. A voltage amplitude of 3.7 volts for typical ODSCC corresponds to this tube burst requirement at a lower 95% prediction interval on the burst correlation. Alternate crack morphologies can correspond to 3.7 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 0.44 inch and 0.76 inch, respectively. Nominal leakage for these crack lengths would range from about 0.2 gpm to 2.3 gpm, respectively, while lower 95% confidence level leak rates would range from about 0.04 gpm to 0.33 gpm, respectively. A leak rate of 150 gpd will provide for detection of 0.4 inch long cracks at nominal leak rates and 0.6 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 criteria at V. C. Summer will help preclude tube burst during all plant conditions during Cycle 8 operation. Tube burst margins are expected to meet or exceed RG 1.121 acceptance criteria. The limiting consequence of the application of the interim plugging criteria is a potential for primary to secondary leakage below the currently allowable value of 1.0 gpm. Based on the previous cycle eddy current results projected to EOC conditions, a maximum SLB leakage of 0.02 gpm is predicted. This amount of leakage would result in off-site radiological consequences well within a small fraction of 10 CFR Part 100 limits. Unacceptable leakage is not anticipated at normal operating or reactor coolant pump locked rotor conditions. Therefore, as the existing tube integrity criteria and 7 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. During the Refueling Cycle 7 outage (projected for March 1993) the eddy current results will be evaluated and a specific leak rate calculation for EOC-8 will be generated. Based on the prior eddy current inspection results, the leak rate is expected to be much less than 1.0 gpm.

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

   The use of the voltage based bobbin probe interim plugging criteria at V. C. Summer has been demonstrated to maintain steam generator tube integrity commensurate with the criteria of Regulatory Guide 1.121. RG 1.121 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 criteria, 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.

It has been determined that the combined effects of LOCA + SSE [loss of coolant accident plus safe shutdown earthquake] on the steam generator component (as required by GDC 2), may result in localized tube deformation in the area of the wedge regions, at the upper support plates.

Analyses results show that for the V. C. Summer steam generators several tubes near wedge locations could be affected in this manner. These tubes have been precluded from application of interim plugging criteria .... 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 expected amount of secondary to primary leakage in the event of a LOCA + SSE is expected to be much less than the maximum primary to secondary leakage associated with the application of this criteria, i.e., 150 gpd per steam generator. Secondary to primary in-leakage 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, since no TSP deflection would be postulated during a LOCA, the presence of the tube support plate is expected to reduce the amount of in-leakage. Any estimation of in-leakage is further reduced based on the expectation that EOC bobbin voltages are predicted (by Monte Carlo) to be a maximum of 2.97 volts, compared to the threshold limit for normal operation leakage of 7.7 volts.

With regard to limits on tube structural integrity as defined by RG 1.121, the proposed V. C. Summer single cycle, 1.0 volt plugging limit provides for additional margin against tube burst at EOC conditions compared to the currently used 40% depth based criteria. Previous studies have indicated that the through-wall growth rate for ODSCC at the TSP's to be about 10 to 15% per cycle. When these values are combined with eddy current uncertainty (assumed to be 15% for depth based calls), an indication just below the plugging limit of 40% can result in EOC reported depths of up to about 70% through-wall. Depth based criteria modeling would permit crack lengths up to the TSP thickness for which 70% depth provides 3 [delta] P capability. This is approximately equal to the structural limit determined by the criteria of RG 1.121. Using the 1.0 volt criteria, a maximum EOC voltage of 2.97 volts is predicted using Monte Carlo. This methodology predicts EOC voltages at a cumulative probability of approximately 99.5% for 200 indications in the population. The EOC voltage which corresponds with a burst pressure equal to three times the normal operating pressure differential is approximately 3.7 volts, therefore, a 20% margin (0.73 volts) between maximum projected EOC voltages and the structural limit voltage is afforded using the voltage based criteria, where little or no margin could be expected for a similar condition tube using the current 40% depth based criteria. Also, comparing the maximum projected EOC voltage with the maximum bobbin voltage obtained during the last inspection indicates that EOC voltages are approximately equal for both criteria. The maximum voltage from the last inspection was 2.81 volts, with a growth of 2.01 volts. The maximum EOC voltage predicted by Monte Carlo of 2.97 volts assumed an initial BOC voltage of 1.0, indicating total growth and uncertainty of 1.97 volts.

Addressing RG 1.83 considerations, implementation of the bobbin probe voltage based interim tube plugging criteria of 1.0 volt is supplemented by: use of a probe wear standard, 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. Whether a depth based or voltage based criteria is used, the peak-to-peak voltage is independent of the phase angle calibration for depth and will inherently provide for a more accurate methodology for assessing degraded tube operability. Pulled tube experience, for some crack morphologies has shown that as crack depth and length increase, bobbin voltage increases also.

As noted previously, implementation of the tube support plate elevation plugging criteria will decrease the number of tubes which must be repaired by sleeving or taken out of service by plugging. The installation of steam generator tube plugs reduces the RCS [reactor coolant system] flow margin. Thus, implementation of the interim plugging criteria 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.

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 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 29, 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 Fairfield County Library, Garden and Washington Streets, Winnsboro, South Carolina, 29180. 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 Jocelyn A. Mitchell: 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 Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina, 29218, 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 February 12, 1993, 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 Fairfield County Library, Garden and Washington Streets, Winnsboro, South Carolina, 29180.

Dated at Rockville, Maryland, this 22nd day of February 1993.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Wunder

George F. Wunder, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation