October 24, 1983

Docket No. 50-259

Mr. Hugh G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

Re: Browns Ferry Nuclear Plant, Unit No. 1

The Commission has filed the enclosed "Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing" with the Office of the Federal Register for publication. The notice is in response to your application dated July 13, 1983, as supplemented July 21, 1983, related to the forthcoming Cycle 6 core reload and plant modifications being performed during the refueling outage.

Sincerely,

Original signed by/

Richard J. Clark, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosure:
Notice

cc w/enclosure:
See next page

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UNITED STATES NUCLEAR REGULATORY COMMISSION

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND 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. DPR-33, issued to Tennessee Valley Authority (the licensee), for operation of the Browns Ferry Nuclear Power Plant, Unit No. 1, located in Limestone County, Alabama.

The amendment would revise the Technical Specifications (T.S.) of the operating license to: 1) modify the core physics, thermal and hydraulic limits to be consistent with the reanalyses associated with replacing about 1/3 of the core during the current refueling outage and 2) reflect plant modifications performed during the current refueling and modification outage, which started on April 16, 1983. Specifically, the amendment would result in changes to the T.S. in the following eleven areas:

1) Changes to the license related to the Cycle 6 core reload involving removal of depleted fuel assemblies in about one-third of the nuclear reactor core and replacement with new fuel of the same type previously loaded in the core with attendant license changes in the core protection safety limits and reactor protection system setpoints. The actual changes are a slight adjustment (by 0.01 in initial core life) in the Operating Limit Minimum Critical Power Ratio (OLMCPR), an added table on maximum average planar linear heat generation rate (MAPLHGR) versus average planar exposure and a change to the bases for the total relief capacity of the safety relief valves.

2) Changes to the T.S. to revise the tables listing surveillance instrumentation for suppression pool bulk temperature reflecting the installation of 16 sensors for an improved torus temperature monitoring system and a revision to the basis for the existing limits on torus water temperature;

3) Changes to the T.S. to reflect modifications to the scram discharge volumes (SDV), and the addition of a second scram discharge instrument volume (SDIV); each of the SDIVs now have redundant vent and drain valves and new, diverse level instrumentation. The changes to the T.S. are to add operability, surveillance and calibration requirements on the new level instrumentation and valves.

4) Changes to T.S. surveillance instrumentation tables to add new instrumentation for containment high-range radiation monitors and to add new instrumentation, and delete current instrumentation for dyrwell pressure-wide range and suppression chamber wide-range water level in response to requirements in NUREG-0737,; items II.F.1.3, II.F.1.4 and II.F.1.5.

5) T.S. changes to incorporate calibration and surveillance requirements for time delay relays to prevent spurious isolation of the HPCI and RCIC systems as required by NUREG-0737; item II.K.3.15.

6) Revision of the T.S. table for containment isolation valve surveillance to add two new isolation valves that are part of a newly installed redundant discharge line from the drywell compressor into containment;

7) Revision of the T.S. to reflect installation of strong backs on personnel airlock doors to allow testing in accordance with 10 CFR Part 50, Appendix J;

8) Revision of T.S. to provide limiting conditions for operation and surveillance requirements for electric power monitoring for the reactor protection system power supply;

9) Modify the T.S. to apply to the new analog (continuous measuring) instrumentation. The analog instrumentation replaces certain mechanical-type pressure and level switches with a more accurate and more stable electronic transmitter/electronic switch system and will provide improved performance of trip functions for reactor protection system actuation, and containment isolation. The changes to the T.S. include:

a. in the tables on functional test frequencies, calibration frequencies and surveillance requirements, for each switch replaced, add the instrument number and type of sensor beneath the parameter being monitored and/or controlled.

b. add notes to the above tables to specify how the functional and calibration tests are to be conducted.

c. in addition to the above administrative changes, the calibration requirements have been changed to incorporate extended calibration intervals. However, the required setpoints, functional test frequencies and channel check frequencies for the instrumentation will not be changed. The new calibration requirements, together with the new instrumentation, are expected to provide a more reliable instrumentation system.

10) Change the T.S. to reflect the addition of a thermal power monitor. The purpose of this monitor is to have the Average Power Range Monitor (APRM) flow biased neutron flux signal respond to the thermal flux rather

than the neutron flux in the core by accounting for the approximately 6-second thermal time constant of the fuel. The specific changes to the T.S. are:

a. Add the words "flow biased" in parenthesis to the heading to the heading for the limits on "APRM Flux Scram Trip Settings" to indicate that the settings are reduced according to the equations given in this section when there is less than 100% core flow.

b. There is a trip unit-separate from the APRM flow-biased scram at less than 120% instantaneous neutron flux. A new requirement is being added to require that whenever the mode switch is in the run position, the APRM fixed high flux scram trip setting shall be operable and set at S 120% power.

c. The bases for the neutron flux scram are revised to describe the functions of the APRM Flow-Biased High Flux Scram Trip Setting and the Fixed High Neutron Flux Scram Trip.

d. Since there is now a new trip system, the tables listing the operability requirements and functional test frequencies on the scram instrumentation have to be revised to add this new instrumentation.

11) Administrative changes to the T.S. involving changes to the Table of Contents to reflect the above license changes, an editorial change and corrections to the list of sample valves to be consistent with present plant configuration.

These revisions to the Technical Specifications would be made in response to the licensee's application dated July 13, 1983, as supplemented July 21, 1983.

Before issuance of the proposed license amendment, the Commission will have made findings as 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.

The Commission has provided guidance concerning the application of the standards by providing examples of actions that are likely, and are not likely, to involve significant hazards considerations (48 FR 14870). The first three examples of actions not likely to involve significant hazards considerations are:

"(i) A purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature.

(ii) A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more stringent surveillance requirement.

(iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable."

Each of the eleven changes to the T.S. described previously is encompassed by one of the above examples of actions not likely to involve a significant hazards consideration. The basis for the staff's determination on each of the eleven changes is discussed below.

1. <u>Core Reload</u>

The changes to the T.S. associated with removing depleted spent fuel from the reactor and replacing these with new fuel assemblies is encompassed by example (iii) above of those actions not likely to involve a significant hazards consideration.

The proposed reload involves fuel assemblies of the same type as previously found acceptable by the staff and loaded in the core in previous cycles. The analytical methods used by the licensee to demonstrate conformance to the technical specifications have been previously approved by the staff. In addition, no changes have been made to the acceptance criteria for the technical specification changes involved.

Since the replacement fuel assemblies are of the same type previously added to all three Browns Ferry units and other BWRs and since the codes, models and analytical techniques used to analyze the reload have been generically approved by the NRC, the changes to the T.S. associated with the reload are clearly encompassed by example (iii) of the guidance provided by the Commission for an action not likely to involve a significant hazards consideration.

2. <u>Changes Related to Torus Modifications</u>

On January 19, 1982, the Commission issued an Order in the matter of Browns Ferry Unit 1 requiring completion - during the current refueling outage - of the plant modification required by the Mark I program so as to comply with the Staff's Acceptance Criteria contained in Appendix A to NUREG-0661. Numerous modifications are being implemented in the Unit 1 torus during the reload 5 refueling outage as part of the Mark I Containment Program. These modifications are required by NRC to restore the originally intended margins of safety in the containment design. Most of the major internal structural modifications to the torus were completed during the previous refueling outage. These modifications are discussed in Amendment No. 76 to Facility Operating License No. DPR-33 issued September 15, 1981. The modifications being made during this outage will complete the requirements specified in NUREG-0661, "Safety Evaluation Report, Mark I Containment Long-Term Program."

One of the changes to the T.S. is to revise the tables that list the surveillance instrumentation associated with the suppression pool bulk temperature. This modification provides an improved torus temperature monitoring system which consists of 16 sensors. This will provide a more accurate indication of the torus water bulk temperature as required by NUREG-0661 and will replace the suppression chamber water temperature instruments presently listed in the T.S.

Another change to the T.S. is to revise the bases for the present limits on temperature of water in the torus. The present bases for suppression pool temperature limits were founded on the Humboldt Bay and Bodega Bay tests. Consistent with the long-term torus integrity program of NUREG-0661 and NUREG-0783, the bases require change to account for steam mass fluxes through the safety/relief valve (S/RV) T-quenchers. The proposed bases describe assurances of stable and complete condensation of steam discharged through the S/RVs and adequate residual heat removal (RHR) and core spray pump net positive suction head.

As noted above, the Commission ordered that the above torus modifications be implemented. The changes to the T.S. are necessary administrative follow up actions essential to the implementation of these improvements. The changes to the T.S. place operability and calibration requirements on the new temperature monitoring system. Since these are new instruments, the surveillance requirements are not presently in the T.S. Thus, adding these restrictions and controls is encompassed by example (ii) provided by the Commission.

The bases for the suppression pool temperature limits are also being changed to account for steam mass fluxes through the safety relief valve T-quenchers as required by NUREG-0661 and NUREG-0783. The proposed bases describe assurances of stable and complete condensation of steam discharged through the S/RVs and adequate RHR and core spray pump net positive suction head. The changes are necessary administrative follow up action essential to the implementation of improvements required by the Commission. Modifying these restrictions is encompassed by example (ii) provided by the Commission.

3. <u>Scram Discharge Instrument Volume</u>

The SDVs and SDIVs are being modified to address inadequacies identified by the partial rod insertion event on Browns Ferry Unit 3 in June 1980. One of the modifications includes adding another valve in series to the existing drain and vent valves on the SDV and SDIV. Another modification includes adding electronic level switches to initiate a scram on a high level in the SDIV. On June 24, 1983, the Commission issued Orders for the Browns Ferry Nuclear Plant, Units 1 and 3 to install permanent Scram Discharge System modifications during the Cycle 5 outages for Units 1 and 3. (This is the Cycle 5 outage for Unit 1.) The modifications have been previously completed for Unit 2. The Orders included "Model Technical Specifications which are provided as guidance for preparing Technical Specification changes that will be required to be approved before operation with the modified system." Both the modification of the systems and submission of T.S. changes to place operability and surveillance requirements on the new instruments and valves were required of the licensee to be in compliance with a Commission Order. Thus, the changes to the T.S. are necessary administrative follow up actions essential to the implementation of these improvements. Adding these new restrictions and controls, which otherwise would not be in the T.S., is encompassed by example (ii) of the guidance provided by the Commission.

4. Accident Monitoring Instrumentation

Item II.F.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," requires all licensees to install five new monitoring systems and to provide onsite sampling/analysis capability for a specified range of radionuclides. For all six categories, NUREG-0737 states: "Changes to technical specifications will be required." During this refueling outage, the licensee has installed: a) a containment high-range monitoring system, b) a drywell wide-range pressure monitoring system and c) a suppression chamber wide-range water level monitoring system. These three items were required by NUREG-0737, items II.F.1.3, II.F.1.4 and II.F.1.5, respectively. The changes to the T.S., which track the model T.S. provided to the licensee by the staff, are to add operability and surveillance requirements on the new monitoring systems to the T.S.

The revisions also delete the present drywell pressure and suppression chamber water level instruments since they are being replaced by items b and c above. The changes to the technical specifications are necessary administrative follow up actions required by the Commission. Adding the new surveillance requirements and controls is encompassed by example (ii) of the guidance provided by the Commission.

5. <u>NUREG-0737, Item II.K.3.15</u>

TMI Action Plan Item II.K.3.15 requires licensees of BWRs to modify pipe-break-detection circuitry so that pressure spikes resulting from high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) initiation will not cause inadvertent system isolation. The licensee elected to employ the BWR Owners Group modification which incorporates a three-second time delay relay (TDR) to prevent spurious isolation. In our letter to the licensee of October 13, 1981, we requested the licensee to provide certain analyses and to "propose the appropriate Surveillance Requirements and Limiting Conditions of Operation for the HPCI and RCIC systems which address this item." The safety evaluation was provided by the licensee's letter of December 16, 1981. All of the Browns Ferry units have had a three-second TDR on the HPCI system. During the current outage for Browns Ferry Unit 1, a TDR was added to the RCIC system. The proposed changes to the Technical Specifications requiring calibration and surveillance of the time delay relays is in accordance with the requirements of NUREG-0737, item II.K.3.15 and the staff's follow up letter. The changes to the T.S. are necessary administrative follow up actions essential to the implementation of these improvements. The addition of requirements in the T.S. for the operability and surveillance of the new time delay relays clearly imposes additional limitations and controls not presently included in the T.S. and is therefore encompassed by example (ii) of the guidance provided by the Commission.

6. Redundant Air Supply to Drywell

During the current outage, TVA has installed a second discharge line from the drywell compressor into containment. This line was added to provide the capability for isolation of approximately one-half of the drywell suppression equipment in the case of a drywell line leak. This air supply will be used to supply two inboard main steam isolation valves (MSIVs), approximately one-half of the main steam relief valves (MSRVs), and approximately one-half of all other air-operated equipment in the drywell. This will significantly reduce the possibility of any one control air pipe break inside containment from requiring immediate shutdown and isolation due to MSIVs, MSRVs, and drywell coolers being inoperable. Since any line penetrating containment requires two isolation valves, the table in the Technical Specifications listing the isolation valves that must be periodically tested is being revised to add these two new isolation valves. TVA has concluded that this modification will increase the margin of safety. The changes to the technical specifications are necessary administrative follow up actions essential to the implementation of this improvement. The two isolation valves being added to the T.S. are new valves not presently listed in the T.S. If they were not added to the table of valves to be periodically tested, there would be no T.S. requirement to test these valves. Adding these additional controls is encompassed by example (ii) of the guidance provided by the Commission.

7. Modification of Airlock Doors

Section III.D.2(b) of Appendix J, 10 CFR Part 50, requires that air locks shall be tested at 6 month intervals at an internal pressure not less than P_a. P_a is defined as "the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases." Reactor plants designed prior to the issuance of Appendix J often do not have the capability to test airlocks at P_a without the installation of strongbacks or the performance of mechanical adjustments to the operating mechanisms of the inner doors. The reason for this is that the inner doors are designed to seat with accident pressure on the containment side of the door, and therefore, the operating mechanisms were not designed to withstand accident pressure in the opposite direction. When the airlock is pressurized for a local airlock test (i.e., pressurized between the doors), pressure is exerted on the airlock side of the inner door, causing the door to unseat and preventing the performance of a meaningful test. The strongback or mechanical adjustments prevent the unseating of the inner door, allowing the test to proceed. Section 4.7.A.2.g of the present T.S. requires that "the personnel air lock shall be tested at a pressure of 49.6 psig during each operating cycle." The proposed change to the T.S. is to require that "the personnel air lock shall be tested at 6-month intervals at an internal pressure of not less than 49.6 psig." This more stringent surveillance requirement is clearly encompassed by example (ii) of the guidance provided by the Commission.

8. Monitoring of RPS Power Supply

By letter dated August 7, 1978, we advised TVA that during review of Hatch Unit 2, the staff had identified certain deficiencies in the design of the voltage regulator system of the motor generator sets which supply power to the reactor protection system (RPS). Pursuant to 10 CFR 50.54(f), TVA was required to evaluate the RPS power supply for Browns Ferry 1, 2 and 3 in light of the information set forth in our letter. Based on our review of TVA's response, by letter dated September 24, 1980, we informed TVA (and most other BWRs) that "we have determined that modifications should be performed to provide fully redundant Class IE protection at the interface of non-Class IE power supplies and the RPS." We also advised TVA that "we have found that the conceptual design proposed by the General Electric Company and the installed modification on Hatch are acceptable solutions to our concern." By letter dated December 4, 1980, TVA committed to install the required modifications. By letters dated October 30, 1981 and July 28, 1982, we sent TVA model Technical Specifications for electric power monitoring of the RPS design modification. During the current outage of Unit 1, the RPS is being modified to provide a fully redundant Class IE protection at the interface of the non-Class IE power supplies and the RPS. This will ensure that failure of a non-Class IE reactor protection power supply will not cause adverse interaction to the Class IE reactor protection system. The Technical Specifications are being revised similar to the model T.S. provided to TVA to reflect the limiting conditions for operation and surveillance requirements associated with the RPS modifications. Page 42 is being modified to add a description of these sections in the bases. The changes to the T.S. are necessary administrative follow up actions essential to the implementation of these improvements. The additional limitations and controls, which are presently not in the T.S., are encompassed by example (ii) of the guidance provided by the Commission.

9. Analog Trip System

The RPS, the primary containment isolation system (PCIS), and the core standby cooling systems (CSCS) use mechanical-type switches in the sensors that monitor plant process parameters. These mechanical-type switches are very subject to drift in the set-point as is evident from the many licensee event reports (LERs) that have been submitted reporting calibration drifts in these switches.

Advances in technology make it possible to replace the mechanical-type switches with a more accurate and more stable electronic transmitter/electronic switch system. For several years, TVA has been planning to replace existing pressure switches that sense drywell and reactor pressures with analog loops and modify the reactor water level indication loops to improve the reliability, accuracy and response time of this instrumentation. The modification involves removing one device and substituting other devices to perform the same function. Changes in design bases, protective function, redundancy, trip point and logic are not involved. Similar modifications have been approved for other BWRs. As described previously, most of the changes to the T.S. are administrative in nature (i.e., adding the specific number and types of sensor and adding notes to describe how testing is conducted). As such, they are encompassed by example (i) of the guidance provided by the Commission. The changes in surveillance requirements relate to example (ii) of the guidance provided by the Commission. Some of the surveillance intervals have been decreased as appropriate for each new instrument. However, the overall effect of the changes in technical specifications will be to increase the total surveillance requirements in support of a more reliable instrumentation system.

10. Thermal Power Monitor

During this outage, the licensee is installing a flow-biased simulated thermal power monitor. These monitors are installed on most all BWRs; the justification for these monitors is discussed in the "Bases" for the APRM settings in the BWR Standard Technical Specifications, NUREG-0123 (BWR/4, STS, Section 2.2.1, page B2-7). The monitors are installed to have the APRM flow-biased neutron flux signal respond to the thermal flux rather than the neutron flux by accounting for the approximately 6-second thermal time constant of the fuel. The addition of the thermal power monitor will prevent a flow-biased neutron flux scram when a transient-induced neutron flux spike occurs that is a short time duration and does not result in an instantaneous heat flux in excess of transient limits. Neutron flux is damped by approximately a 6-second fuel time constant. This feature will reduce the number of scrams due to small fast flux transients such as those which result from control valve and MSIV testing and small perturbations in water level and pressure.

A thermal power monitor was installed in Browns Ferry Unit 2 during the last outage and approved by Amendment No. 85 to Facility Operating License No. DPR-52 issued March 11, 1983.

As identified previously, the changes to the T.S. are to add operability and functional test frequency requirements for this new trip system and to add a description of this new trip system in the "Bases." The changes to the T.S. are necessary administrative follow up actions essential to the implementation of these improvements. The additional limitations, restrictions and controls, which are not presently included in the T.S., are encompassed by example (ii) of the guidance provided by the Commission.

11. Administrative Changes

Several administrative changes are being made to the Technical Specifications. These include revising the Table of Contents to reflect the changes discussed above, an editorial change and corrections to the list of sample valves to reflect the current plant configuration. These changes are editorial in nature and have no safety significance. These changes are encompassed by example (i) cited by the Commission as an action not likely to pose a significant hazards consideration.

Since all of the changes to the T.S. are encompassed by an example in the guidance provided by the Commission of actions not likely to involve a significant hazards consideration, the staff has made a proposed determination that the application for amendment 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.

Comments should be addressed to the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attn: Docketing and Service Branch.

By November 28, 1983, 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 petition for leave to intervene. Request for a hearing and petitions 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. 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, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. 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 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 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 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, D. C. 20555, Attn: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., 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 (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to Domenic B. Vassallo: 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 Executive Legal Director, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, and to H. S. Sanger, Jr., Esquire, General Counsel, Tennessee Valley Authority, 400 Commerce Avenue, E11B 33C, Knoxville, Tennessee 37902, 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 Atomic Safety and Licensing Board designated to rule on the petition and/or request, that the petitioner has made a substantial showing of good cause for the granting of a late petition and/or request. That determination will be 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 July 13, 1983, as supplemented July 21, 1983, which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Athens Public Library, South and Forrest, Athens, Alabama 35611.

Dated at Bethesda, Maryland, this 24th day of October, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

DMVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing