July 26, 1995

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSE. PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION, AND OPPORTUNITY FOR HEARING - SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M92961 AND M92962)

Dear Mr. Kingsley:

Enclosed is a copy of the subject notice for your information. This notice relates to your application dated July 19, 1995, to amend the Sequoyah Nuclear Plant, Units 1 and 2 Technical Specifications to incorporate new requirements associated with steam generator tube inspections and repair. The new requirements would establish alternate steam generator tube plugging criteria at the tube support plate intersections.

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

Sincerely,

Original signed by

David E. LaBarge, Sr. Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 328

Enclosure:
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Mr. Oliver D. Kingsley, Jr.
Tennessee Valley Authority

SEQUOYAH NUCLEAR PLANT

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UNITED STATES NUCLEAR REGULATORY COMMISSION

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 328

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

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License Nos. DPR-77 and DPR-79 issued to the Tennessee Valley Authority (the licensee) for operation of the Sequoyah Nuclear Plant, Units 1 and 2, located in Soddy Daisy, Tennessee.

The proposed amendments would amend the Sequoyah Nuclear Plant, Units 1 and 2 Technical Specifications to incorporate new requirements associated with steam generator tube inspections and repair. The new requirements would establish alternate steam generator tube plugging criteria at the tube support plate intersections.

Before issuance of the proposed license amendments, 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

ENCLOSURE

facility in accordance with the proposed amendments 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:

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

1. 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 shows burst pressures in excess of 5,000 pounds per square inch (psi) for indications of outer diameter stress corrosion cracking with voltage measurements as high as 19 volts. Burst testing performed on intersections pulled from SQN with up to a 1.9-volt indication shows measured burst pressure in excess of 6,600 psi at room temperature. Burst testing performed on pulled tubes from other plants with up to 7.5-volt indications shows burst pressures in excess of 5,200 psi at room temperatures. 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 safety-factor requirements of NRC Regulatory Guide (RG) 1.121.

   Tube burst criteria are inherently satisfied during normal operating conditions because of the proximity of the tube support plate (TSP). Since tube-to-tube support plate proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics that maintain a margin of safety of 1.43 times the bounding faulted condition steam line break (SLB) pressure differential. During a postulated SLB, the TSP has the potential to deflect during blowdown following a main SLB, thereby uncovering the TSP intersections.

   Based on the existing database, the RG 1.121 criterion requiring maintenance of a safety factor of 1.43 times the

SLB pressure differential on tube burst is satisfied by 7/8-inch-diameter tubing with bobbin coil indications with signal amplitudes less than 8.82 volts (WCAP-13990), regardless of the indicated depth measurement. A 2.0-volt plugging criterion (resulting in a projected end-of-cycle [EOC] voltage) compares favorably with the 8.82-volt structural limit considering the extremely slow apparent voltage growth rates and few numbers of indications at SQN. Using the established methodology of RG 1.121, the structural limit is reduced by allowances for uncertainty and growth to develop a beginning of cycle (BOC) repair limit that would preclude indications at EOC conditions that exceed the structural limit. The nondestructive examination (NDE) uncertainty component is 20.5 percent, and is based on the Electric Power Research Institute (EPRI) alternate repair criteria (ARC).

Test data indicates that tube burst cannot occur within the TSP, even for tubes that have 100 percent throughwall electro-discharge machining notches, 0.75 inch long, provided that the TSP is adjacent to the notched area. Because of the few number of indications at SQN, the EPRI methodology of applying a growth component of 35 percent per effective full power year (EFPY) will be used. Near-term operating cycles at SQN are expected to be bounded by 1.23 years, therefore, a 43 percent growth component is appropriate. When these allowances are added to the BOC alternate plugging criteria (APC) of 2.0 volts in a deterministic bounding EOC voltage of approximately 3.26 volts for Cycle 7, operation can be established. A 5.56-volt deterministic safety margin exists (8.82 structural limit - 3.26-volt EOC equal 5.56-volt margin).

For the voltage/burst correlation, the EOC structural limit is supported by a voltage of 8.82 volts. Using this structural limit of 8.82 volts, a BOC maximum allowable repair limit can be established using the guidance of RG 1.121. The BOC maximum allowable repair limit should not permit the existence of EOC indications that exceed the 8.82-volt structural limit. By adding NDE uncertainty allowances and an allowance for crack growth to the repair limit, the structural limit can be validated. Therefore, the maximum allowable BOC repair limit (RL) based on the structural limit of 8.82 volts can be represented by the expressions:

$$RL + (0.205 \times RL) + (0.43 \times RL) = 8.82 \text{ volts, or,}$$

the maximum allowable BOC repair limit can be expressed as,

$$RL = 8.82\text{-volt structural limit}/1.64 = 5.4 \text{ volts.}$$

This RL (5.4 volts) is the appropriate limit for APC implementation to repair bobbin indications greater than 2.0 volts independent of rotating pancake coil (RPC) confirmation of the indication. This 5.4-volt upper limit for non-confirmed RPC calls is consistent with other recently approved APC programs (Farley Nuclear Plant, Unit 2).

The conservatism of the growth allowance used to develop the repair limit is shown by the most recent SQN eddy current data. Two tubes plugged in Unit 1 during the last outage had less than one volt of growth over the past five operating cycles. Only seven tubes in Unit 2 required repair because of outside diameter stress corrosion cracking (ODSCC) at the TSP intersections.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated main SLB outside of containment, but upstream of the main steam isolation valve (MSIV), represents the most limiting radiological condition relative to the APC. Implementation of the APC will determine whether the distribution of cracking indications at the TSP intersections is 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. A separate analysis has determined this allowable SLB leakage limit to be 4.3 gallons per minute (gpm) in the faulted loop. This limit uses the TS reactor coolant system (RCS) Iodine-131 activity level of 1.0 microcuries per gram dose equivalent Iodine-131 and the recommended Iodine-131 transient spiking values consistent with NUREG-0800. The analysis method is WCAP-14277, which is consistent with the guidance of the NRC draft generic letter (GL) and will be used to calculate EOC leakage. Because of the relatively low number of indications at SQN, it is expected that the actual leakage values will be far less than this limit. Additionally, the current Iodine-131 levels at SQN range from about 25 to 100 times less than the TS limit.

Application of the criteria requires the projection of postulated SLB leakage, based on the projected EOC voltage distribution for Cycle 8 operation. Projected EOC voltage distribution is developed using the most recent EOC eddy current results and a voltage measurement uncertainty. Data indicates that a threshold voltage of 2.8 volts would result in throughwall cracks long enough to leak at SLB condition. The draft GL requires that all indications to which the APC are applied must be included in the leakage projection. Tube pull results from another plant with 7/8-inch tubing with a substantial voltage growth database have shown that tube wall degradation of greater than 40 percent throughwall was readily detectable either by the bobbin or RPC probe.

The tube with maximum throughwall penetration of 56 percent (42 average) had a voltage of 2.02 volts. The SQN Unit 1 pulled tube had a 1.93-volt indication with a maximum depth of 91 percent and did not leak at SLB condition. Based on the SQN pulled tube and industry pulled tube data supporting a lower threshold for SLB leakage of 2.8 volts, inclusion of all APC intersections in the leakage model is quite conservative. The ODSCC occurring at SQN is in its earliest stages of development. The conservative bounding growth estimations to be applied to the expected small number of indications for the upcoming inspection should result in very small levels of predicted SLB leakage. Historically, SQN has not identified ODSCC as a contributor to operational leakage.

In order to assess the sensitivity of an indication's BOC voltage to EOC leakage potential, a Monte Carlo simulation was performed for a 2.0-volt BOC indication. The maximum EOC voltage (at 99.8 percent cumulative probability) was found to be 4.8 volts. The leakage component from an indication of this magnitude, using either the NUREG-1477 or EPRI leakage models, is 0.12 or 0.028 gpm, respectively.

Therefore, as implementation of the 2.0-volt APC does not adversely affect steam generator (S/G) tube integrity and implementation will be shown to result in acceptable dose consequences, the proposed amendment does not result in significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

   Implementation of the proposed S/G tube APC does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism that could result in an accident outside of the region of the TSP elevations; no ODSCC is occurring outside the thickness of the TSP. Neither a single or multiple tube rupture event would be expected in a S/G in which the plugging criteria is applied (during all plant conditions).

   TVA will implement a maximum leakage rate limit of 150 gallon per day per S/G to help preclude the potential for excessive leakage during all plant conditions. The SQN TS limits on primary-to-secondary leakage at operating conditions include a maximum of 0.42 gpm (600 gallons per day [gpd]) for all S/Gs, or, a maximum of 150 gpd for any one S/G. The RG 1.121 criterion for establishing operational leakage rate limits that require plant shutdown is based upon leak-before-break considerations to detect a free-span crack before potential tube rupture during faulted plant conditions. 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. 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 length that provides a factor of safety of 1.43 against bursting at faulted conditions maximum pressure differential. A voltage amplitude of 8.82 volts for typical ODSCC corresponds to meeting this tube burst requirement at a lower 95 percent prediction limit on the burst correlation coupled with 95/95 lower tolerance limit material properties. Alternate crack morphologies can correspond to 8.82 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 1.43 times the SLB pressure differential and the SLB pressure differential alone are approximately 0.57 inch and 0.84 inch, 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 percent confidence level leak rates. Since tube burst is precluded during normal operation because of the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during SLB conditions, the leakage from the maximum permissible crack must preclude tube burst at SLB conditions. Thus, the 150-gpd limit provides for plant shutdown before reaching critical crack lengths for SLB conditions. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncover will provide benefit to the burst capacity of the intersection.

As S/G tube integrity upon implementation of the 2.0-volt APC continues to be maintained through in-service inspection and primary-to-secondary leakage monitoring, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Involve a significant reduction in a margin of safety.

The use of the voltage based APC at SQN is demonstrated to maintain S/G tube integrity commensurate with the criteria of RG 1.121. RG 1.121 describes a method acceptable to the NRC Staff for meeting General Design Criteria (GDC) 14, 15, 31, and 32 by reducing the probability or the consequences of S/G tube rupture. This is accomplished by determining the

limiting conditions of degradation of S/G tubing, as established by in-service 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 TSP elevations is not expected to lead to a S/G tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the TSP elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions and radiological consequences are not adversely impacted.

In addressing the combined effects of loss-of-coolant accident (LOCA), plus safe shutdown earthquake (SSE) on the S/G component (as required by GDC 2), it has been determined that tube collapse may occur in the S/Gs at some plants. This is the case as the TSP may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate because of 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 S/G tube collapse. First, the collapse of S/G 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.

Consequently, since the leak-before-break methodology is applicable to the SQN reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes either the accumulator line break or the pressurizer surge line break. LOCA loads for the primary pipe breaks were used to bound the conditions at SQN for smaller breaks. The results of the analysis using the larger break inputs show that the LOCA loads were found to be of insufficientmagnitude to result in S/G tube collapse or significant deformation. The LOCA, plus SSE tube collapse evaluation performed for another plant with Series 51 S/Gs using bounding input conditions (large-break loadings), is applicable to SQN. Therefore, at SQN, no tubes will be excluded from using the voltage repair criteria due to deformation of collapse of S/G tubes following a LOCA plus an SSE.

> Addressing RG 1.83 considerations, implementation of the bobbin probe voltage based interim tube plugging criteria of 2.0 volt is supplemented by: (1) enhanced eddy current inspection guidelines to provide consistency in voltage normalization, (2) a 100 percent eddy current inspection sample size at the TSP elevations, and (3) RPC inspection requirements for the larger indications left in service to characterize the principal degradation as ODSCC.
>
> As noted previously, implementation of the TSP elevation plugging criteria will decrease the number of tubes that must be repaired. The installation of S/G tube plugs reduces the RCS flow margin. Thus, implementation of the alternate 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 of safety.

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.

Normally, the Commission will not issue the amendments 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 amendments involve 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.

Written comments may be submitted by mail to the Rules Review and 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 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, 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.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By August 31, 1995 , 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, and at the local public document room located at the Chattanooga-

Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402. 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.

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, 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 Frederick J. Hebdon: 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 General Council, Tennessee Valley Authority, ET 11H, 400 West Summit Hill Drive, 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 ntertained 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 July 19, 1995, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the local public document room located at the Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

Dated at Rockville, Maryland, this 26th day of July 1995

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

David E. LaBarge, Sr. Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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