December 11, 2006

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT REGARDING ALTERNATE ROD POSITION MONITORING METHOD (TAC NOS. MD2514 AND MD2515) (TS-06-04)

Dear Mr. Singer:

The Commission has issued the enclosed Amendment No. 315 to Facility Operating License No. DPR-77 and Amendment No. 304 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2. These amendments are in response to your application dated July 6, 2006 (TS-06-04), "Monitoring of Control or Shutdown Rod Position by an Alternate Means."

The amendments revise Action a of Technical Specification Limiting Condition for Operation 3.1.3.2, "Position Indication Systems - Operating," to allow for the use of an alternate means other than the movable incore detectors to monitor the position of a control or shutdown rod should problems occur with the analog rod position indication system.

A copy of the safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/
Douglas V. Pickett, Senior Project Manager
Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-327 and 50-328

Enclosures:
1. Amendment No .315 to License No. DPR-77
2. Amendment No. 304 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

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Tennessee Valley Authority

SEQUOYAH NUCLEAR PLANT

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.315
License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Tennessee Valley Authority (the licensee) dated July 6, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

    (2) <u>Technical Specifications and Environmental Protection Plan</u>

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 315 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, and shall be implemented no later than 45 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Evangelos C. Marinos for

Douglas V. Pickett, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. DPR-77
and the Technical Specifications

Date of Issuance: December 11, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 315

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace page 3 of Operating License No. DPR-77 with the attached page 3.

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 1-17	3/4 1-17
3/4 1-17a	3/4 1-17a

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis, instrument calibration
or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation
of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 315 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program
(set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at power level different from there described; and

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 304
License No. DPR-79

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Tennessee Valley Authority (the licensee) dated July 6, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 304 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, and shall be implemented no later than 45 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Evangelos C. Marinos for

Douglas V. Pickett, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. DPR-79
and the Technical Specifications

Date of Issuance: December 11, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 304

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace page 3 of Operating License No. DPR-79 with the attached page 3.

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 1-17	3/4 1-17
3/4 1-17a	3/4 1-17a

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 304 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program
(set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at power level different from there described; and

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 315 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 304 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By application dated July 6, 2006 (ML061990487 [Agencywide Documents Access and Management System Accession Number]), the Tennessee Valley Authority (TVA, the licensee) requested changes to the Technical Specifications (TSs) for the Sequoyah Nuclear Plant (SQN), Units 1 and 2. The requested changes would revise Action a of TS Limiting Condition for Operation (LCO) 3.1.3.2, “Position Indication Systems - Operating,” to allow for the use of an alternate means other than the movable incore detectors to monitor the position of a control or shutdown rod should problems occur with the analog rod position indication (ARPI) system. The use of this alternate method will reduce the frequency of flux mapping using the movable incore detectors to determine the position of the non-indicating rod. This will reduce the wear on the movable incore detector system that is also used to complete other required TS surveillances.

Notice of these amendments was published in the *Federal Register* on August 15, 2006 (71 FR 46938).

2.0 REGULATORY EVALUATION

The objectives of the rod control system and rod position indication system are to ensure that control rod alignment and insertion limits are maintained. Operators utilize the ARPI system to monitor the positions of the rods to establish that the plant is operating within the bounds of the accident analysis assumptions. Operability, including position indication, of the control rods and shutdown rods is an initial condition assumption in all safety analyses that assume rod insertion upon a reactor trip. Maximum rod misalignment is an initial condition assumption in the safety analysis that directly affects core power distributions and assumptions of available shutdown margin (SDM). Control rod inoperability or misalignment may cause increased power peaking due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown.

General Design Criterion (GDC) 13 in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, specifies that instrumentation shall be provided to monitor variables and systems over their operating ranges during normal operation, anticipated operation occurrences, and accident conditions. SQN TS 3.1.3.2 requires operability of the shutdown and control rod ARPI system and the bank demand position indication system, and thereby ensures compliance with the control rod alignment and insertion limits. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the licensee's license amendment request (LAR) to verify that the licensing basis criteria stated in the Updated Final Safety Analysis Report (UFSAR) continues to be met with the proposed changes.

3.0 TECHNICAL EVALUATION

The licensee proposed the following changes to TS 3.1.3.2, Action a:

- Existing Action a.2 will be renumbered to become Action a.3. This is an editorial change which is acceptable to the NRC staff.

- Introduce a new Action a.2 as the alternate means for rod position monitoring. The proposed alternate means described in Action a.2 is composed of three elements; a.2.a), a.2.b), and a.2.c), as follows:

  2. a) Determine the position of the non-indicating rod indirectly by the movable incore detectors within 8 hours and once every 31 days thereafter and within 8 hours if rod control system parameters indicate unintended movement, and

     b) Review the parameters of the rod control system for indications of unintended rod movement for the rod with an inoperable position indicator within 16 hours and once per 8 hours thereafter, and

     c) Determine the position of the non-indicating rod indirectly by the movable incore detectors within 8 hours if the rod with an inoperable position indicator is moved greater than 12 steps and prior to increasing THERMAL POWER above 50% RATED THERMAL POWER and within 8 hours of reaching 100% RATED THERMAL POWER, or

- Add the following footnote regarding the proposed alternate method a.2:

  Rod position monitoring by Actions 2.a), 2.b), and 2.c) may only be applied to one inoperable rod position indicator and shall only be allowed: (1) until the end of the current cycle, or (2) until an entry into MODE 5 of sufficient duration, whichever occurs first, when the repair of the inoperable rod position indication can safely be performed. Actions 2.a), 2.b), and 2.c) shall not be allowed after the plant has been in MODE 5 or other plant condition, for a sufficient period of time, in which the repair of the inoperable rod position indication could have safely been performed.

Action a.2.a) includes three distinct requirements for verification of the non-indicating rod position using the movable incore detectors: (1) initial verification within 8 hours of the inoperability of the ARPI, (2) re-verification once every 31 days thereafter, and (3) verification within 8 hours if rod control system parameters indicate unintended rod movement. The licensee defines an unintended rod movement as the release of the stationary gripper when no action was demanded either manually or automatically from the rod control system, or a rod motion in a direction other than the direction demanded by the rod control system. (An intended rod movement occurs when either an operator manually demands motion from the rod control system, or a temperature or power mismatch demands motion while the rod control system is being controlled automatically.) Any indication of unintended rod movement will be treated as if the ARPI had initially failed, and therefore requires verification of the rod position using the movable incore detectors within 8 hours. The 8-hour completion time for initial verification of rod position is more restrictive than the 12-hour completion time required by the existing method using the movable incore detector specified in Action a.1. Regarding the 31-day verification time interval, the licensee cites the concern over the potential wear and failure of the movable incore detector system caused by repeated use over a long period of time, and therefore the 31-day interval is chosen to coincide with the frequency of other surveillance requirements using the movable incore detectors. The 31-day verification interval is also supplemented by Action a.2.b), using an alternate means to monitor the parameters of the rod control system for indications of rod movement.

Action a.2.b) uses an alternate means to monitor whether the non-indicating rod has moved from the position last determined using the movable incore detectors within 16 hours and once per 8 hours thereafter. The July 6, 2006, LAR provided a detailed description of the alternate method, including components relied upon in the alternate monitoring method. The alternate method will monitor the rod control system parameters for indications of rod movement for the non-indicating rod through a resistor pack. The monitoring of rod control movement consist of monitoring the affected control rod drive mechanism (CRDM) lift coil current, the stationary gripper coil current, and the rod-in and rod-out demands by the plant process computer. The lift coil current and the stationary gripper coil current will be monitored at the rod control cabinet using the permanently installed test points that monitor the voltage drop across a temporarily installed metering resistor. No change in the voltage or coil current output would indicate that the rod has not moved. These measurement voltage signals will be routed to the process computer analog inputs. Temporary connections will be made between the computer room cable ends and the computer input/output. The digital recorder output located on a control board in the main control room, adjacent to the control board where the displays for the ARPIs are located, will continue to display the step number of the rod location of the non-indicating rod.

The proposed method will provide indication of the rod position as last determined using the movable incore detectors, as well as the ability for a unit operator to continuously monitor the position of the affected rod via a digital recorder. The licensee proposes to implement a software algorithm that will set off different alarms when the rod does not move on demand, or when it moves unintentionally. The functions supported by the software algorithm include numerical display of rod position, rod-to-rod deviation alarm, rod-to-bank deviation alarm, monitoring of rod position, rod movement, and alarm and monitoring capabilities of the proposed alternate plan. The licensee tested the software using the signal data obtained from CRDM timing tests as a means to verify the operation of the software algorithm. The licensee described the different scenarios that will generate a plant computer alarm, via the installed software algorithm, when the alternate method is used to ensure that the position of the rod is known. An alarm requiring operator action will be generated for the following conditions: (a) the rod stepped in the wrong direction, (b) the rod stepped with no demand (whether in automatic or manual control), (c) the alternate monitoring circuit failed, or (d) the affected rod operated outside of the rod-to-rod or rod-to-bank deviation limits. The NRC staff finds that the plant computer has provided necessary information for operators to understand the status of the ARPI system, and has sufficient information for operators to perform required operating procedures. When the rod control system indicates an unintended rod movement or a rod movement exceeding 12 steps, respectively, Action a.2.a) or a.2.c) is invoked to determine the rod position using movable incore detectors within 8 hours.

Action a.2.c) consists of three distinct requirements for verification of non-indicating rod position using the movable incore detectors: (1) when the non-indicating rod is (intentionally or unintentionally) moved greater than 12 steps, (2) prior to increasing thermal power above 50 percent rated thermal power (RTP), and (3) within 8 hours after reaching 100 percent RTP. The first requirement ensures that any unintended rod movement is identified in a timely manner, and that the position of the non-indicating rod is identified during power escalations that result from unit trips or power reductions. The second and third requirements will verify the non-indicating rod position prior to power exceeding 50 percent RTP and after reaching 100 percent RTP. Action a.2.c) confirms the position of the non-indicating rod to ensure that power distribution requirements are not violated and to establish a starting point for the proposed alternate monitoring actions.

The staff evaluated the effect of using the alternate monitoring method on the design basis events of a rod drop or rod misalignment event during power operation. The ability to immediately detect a rod drop or misalignment is not directly provided by the movable incore detectors used in current Action a.1, or by the alternate monitoring method by monitoring the rod control system parameters. However, should there be a full-rod drop of a control or shutdown rod, it will be immediately detectable by means other than the position indication system. Independent indication of a dropped rod is obtained using the excore power range signals. Additionally, a negative reactivity insertion corresponding to the reactivity worth of a full-rod drop will cause a change in core parameters including core average temperature and axial flux. Rod misalignment will also be detectable by other means, such as axial flux deviations or a channel deviation alarm. Therefore, the likelihood of an undetected rod drop or misalignment is considered negligible while using the alternate monitoring method. In addition, the limits of LCO 3.1.3.1 on shutdown or control rod alignment of 12 steps ensure that the assumptions in the safety analyses will remain valid and that the assumed negative reactivity will be available to be inserted during a plant shutdown. Therefore, the NRC staff finds that the design basis analyses of a rod drop or misalignment remain acceptable.

While implementing the Required Actions a.1 (using the movable incore detectors) or a.2 (using the alternate means), the rod bottom indication will not be available for the rod with inoperable ARPI to verify full rod insertion following a reactor trip or shutdown. Therefore, the non-indicating rod should be assumed to be incapable of providing negative reactivity following a reactor trip and should not be credited in the SDM calculation. SQN has procedures in place that address the condition in which more than one rod may not fully insert on a reactor trip to ensure that the reactor is safely shutdown. SDM calculations are performed in accordance with surveillance instruction (SI) 0-SI-NUC-000-038.0. Emergency operating procedures already in place are relied upon to address the SDM requirements following a reactor trip. The non-indicating rod would be treated as if it has not fully inserted on a reactor trip and operators will take actions as currently driven by procedures to safely shutdown the reactor, including initiation of emergency boration if two or more rods are not indicating fully inserted. Therefore, the NRC staff concludes that there are adequate controls to provide reasonable assurance that the plant will continue to achieve subcriticality on a reactor trip.

As a result, the NRC staff concludes that the use of alternate means for monitoring the non-indicating rod position provides an acceptable process for knowing the non-indicating rod position and therefore continues to meet GDC 13.

The footnote to TS LCO 3.1.3.2 Action a.2 clearly limits the conditions for the use of the alternate rod position monitoring provisions. It specifically limits the use of these provisions to only one inoperable rod position indicator, thus ensuring sufficient rod position monitoring on a real time basis to verify core conditions during normal operations and accident conditions. The duration for Action a.2 is limited to the end of the current fuel cycle or an entry into Mode 5 with sufficient duration to repair the ARPI safely. Therefore, the repair of the ARPI must be performed as soon as reasonable conditions exist to safely perform the activities and repeated use of this provision is not acceptable in lieu of the necessary repair.

The licensee will treat the proposed alternate monitoring method as a temporary alteration (TA) in accordance with TVA procedure Standard Programs and Processes 9.5, "Temporary Alterations." A 10 CFR 50.59 screening review is performed whenever the TA is implemented. The planning process for a work order used to implement a TA is the same as that used for a permanent plant modification.

In the application, the licensee stated that they would update the UFSAR to clarify the capabilities provided by the alternate monitoring process. This will be accomplished through a revision to Section 7.7.1.3.2, "Rod Position Monitoring of Full Length Rods," of the UFSAR. The proposed revision will discuss the rod control system monitoring process and will clarify that while the alternate monitoring is in use, the operation of the system will be periodically verified through the implementation of Surveilance Requirements (SRs) 4.1.3.1.2. This requires verification of rod trip by demonstrating rod freedom of movement every 92 days. Also, SRs 4.2.2.2 and 4.2.3.3, respectively, require evaluation of heat flux hot channel factor and enthalpy rise factor within their acceptable limits using the power distribution map obtained using the incore detectors.

Based on the evaluation described above, the NRC staff concludes that the proposed TS changes provide adequate controls to ensure that the rod position is known, that any rod misalignment is detectable for the one rod with an inoperable ARPI, and that operators will take appropriate action to ensure that the rod stays within its alignment limit and SDM is maintained. Additionally, the NRC staff agrees that inoperable ARPI should be repaired at the end of the cycle or during an entry into Mode 5 of sufficient duration, when the repair can be safely performed as stated in the Footnote of TS 3.1.3.2.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (71 FR 46938). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Yi-Hsiung Hsii

Date: December 11, 2006