Docket Nos. 50-327
and 50-328

March 15, 1994

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

Dear Mr. Kingsley:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M85308 AND M85309) (TS 92-16)

The Commission has issued the enclosed Amendment No. 177 to Facility Operating License No. DPR-77 and Amendment No. 168 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. These amendments are in response to your application dated January 8, 1993; which was supplemented by submittals dated April 1, May 3, and August 18, 1993; and February 22, 1994.

The amendments remove the surveillance requirement to perform reactor vessel nozzle inspections at the end of each 10-year inspection interval subject to changes in the Sequoyah Unit 1 and 2 Inservice Inspection (ISI) programs to include the commitments to perform the augmented ISI examinations described in TVA's letter of February 22, 1994.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:
David E. LaBarge, Sr. Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 177 to License No. DPR-77
2. Amendment No. 168 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures:
See next page
*see previous concurrence

NAME:	PDII-4/LA	PDII-4/PM	EMCB*	OGC*	PDII-4/D
OFFICE:	BClayton	DLaBarge	JStrosnider	AJorgensen	FHebdon
DATE:	3/14/94	3/14/94	3/10/94	3/10/94	3/15/94

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

SEQUOYAH NUCLEAR PLANT

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AMENDMENT NO. 177 FOR SEQUOYAH UNIT NO. 1 - DOCKET NO. 50-327 and
AMENDMENT NO. 168 FOR SEQUOYAH UNIT NO. 2 - DOCKET NO. 50-328
DATED: March 15, 1994

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cc: Plant Service List

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 177
License No. DPR-77

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated January 8, 1993; which was supplemented by submittals dated April 1, May 3, and August 18, 1993; and February 22, 1994, 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 177, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 15, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 177

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Revise the Appendix A Technical Specifications by removing the page identified below and inserting the enclosed page. The revised page contains a marginal lines indicating the area of change.

REMOVE	INSERT
3/4 4-27	3/4 4-27

REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 AND 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10. The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10. In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C 4.b of Regulatory Guide 1.14, Revision 1, August 1975.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
License No. DPR-79

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated January 8, 1993; which was supplemented by submittals dated April 1, May 3, and August 18, 1993; and February 22, 1994, 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

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 168, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 15, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 168

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Revise the Appendix A Technical Specifications by removing the page identified below and inserting the enclosed page. The revised page contains a marginal lines indicating the area of change.

REMOVE	INSERT
3/4 4-32	3/4 4-32

REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 177 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO.168 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 January 8, 1993; which was supplemented by submittals dated April 1, May 3, and August 18, 1993; and February 22, 1994, the Tennessee Valley Authority (the licensee) proposed amendments to the Technical Specifications (TS) for Sequoyah Nuclear Plant (SQN) Units 1 and 2. The requested changes would remove Surveillance Requirement (SR) 4.3.10.b from the TS. This SR presently requires performance of an inspection of the reactor vessel nozzles at the end of each 10-year inspection interval using techniques at least as sensitive as those used to conduct the supplemental examination performed prior to fuel loading and submitting the results of the examination to the Commission.

The supplemental letters listed above supplied clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 DISCUSSION

The RPV nozzles are manufactured from steel with a stainless steel cladding on the inside surface. The cladding is welded to the steel with a single-pass or multiple-pass welding process. In 1971, underclad cracks were identified in RPV nozzles located in Europe and, subsequently, were addressed in Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." Since the late 1970s, underclad cracks have been basically classified as either reheat or cold cracks.

Reheat cracks, sometimes called stress-relief cracks, are associated with the single-pass-weld cladding process. Single-pass-weld cladding was the process used on Unit 2. In this process, reheat cracks occur during post-weld heat treatment for stress relief. The cracks are confined to the coarse-grain area of the heat-affected zone (HAZ) at prior austenite grain boundaries of nozzles manufactured from American Society and Mechnical Engineers (ASME) SA-508 Class 2 material or similar material. The cracks are located in the weld overlap area between passes and are perpendicular to the direction in which the beads were laid down. The short, shallow cracks are embedded below the

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exposed surface of the cladding. The cracks are influenced by high-heat input during the weld-cladding processes and by small concentrations of residual elements that are located at the austenite grain boundaries. The NRC has accepted for referencing, the conclusions about underclad cracks (now described as reheat cracks) from Topical Report WCAP-7733, "Reactor Vessels, Weld Cladding - Base Metal Interaction." A summary of the conclusions from WCAP-7733 is that reheat cracks are not a safety concern in the cladded area of the RPV nozzles.

Cold cracks, sometimes referred to as hydrogen-induced cracks, are associated with the multiple-pass-weld cladding process that is applied with insufficient preheating to the base material or prior cladded surfaces. Multiple-pass-weld cladding was the process used on Unit 1. The cold cracks are located in the HAZ in the layers of cladding and can also occur in the HAZ of the base metal. The cracks occur from the effects of hydrogen embrittlement in predominantly martensitic, coarse-grain material that is subjected to internal stresses. The short, shallow cracks are perpendicular to the direction in which beads were laid down, and embedded below the surface of the cladding. In the absence of knowing the manufacturing process used for depositing the weld-cladding on the RPV nozzles, indications of suspected cold cracks may be mistaken for suspected reheat cracks.

The staff addressed the testing and evaluation performed in 1980 by Westinghouse and TVA regarding the potential for cold cracking, in NUREG-0011, Supplement 1, "Safety Evaluation Report Related to Operation of Sequoyah Nuclear Plant, Units 1 and 2," dated February 1980. The report documented the tests performed in 1980 and determined that the underclad cracks that were identified were within the acceptance standards contained in IWB-3514-2 of Section XI of the 1977 Edition of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The evaluation is documented in Sections 5.2.6 of the SQN Safety Evaluation Report. In addition, in order to provide added assurance that adequate margins were maintained during service, the NRC required that TVA insert a requirement to perform supplemental examinations in the TS.

The supplemental examinations performed in 1980 consisted of manually operated ultrasonic testing (UT) techniques applied to the inside surface of the RPV nozzles. The supplemental examination performed on Unit 1 during the first 10-year ISI interval was performed in April 1993 and consisted of an automatically operated UT technique with computer enhancements to assist in data analysis.

Following discussions with the staff related to testing and analysis, TVA conducted the nozzle inspections for Unit 1 during the Cycle 6 refueling outage in 1993, and documented the results in a submittal dated August 18, 1993. The sensitivity of the UT technique used in the 1993 supplemental examination had been previously demonstrated in tests conducted at the Southwest Research Institute in April 1993. In the submittal, TVA concluded that the inherent differences between the 1980 and 1993 UT techniques interfered with the reproducibility of the inspection results, making direct comparisons unsuccessful. Instead, TVA concluded that the 1993 supplemental examination would be used as a baseline for Unit 1 comparisons that would be made during the next (second) 10-year ISI interval.

The August 18, 1993, submittal also reiterated TVA's proposal to remove the supplemental examination requirement from the TS. By removing the requirement, both units would be committed to the normal 10-year ISI program using Section XI of the ASME Code, 1977 Edition, Summer 1978 Addenda, as the ISI code of record. In the ISI program, the RPV nozzle examinations are performed in accordance with Examination Table IWB-2500-1, Categories B-D and B-F, Item Numbers B3.90, B3.100, and B5.10.

3.0 EVALUATION

3.1 Unit 1

During the 1993 inspection (completed during the U1C6 refueling outage), TVA identified 46 reflectors in 7 of the 8 nozzles. TVA detected 6 new reflectors and could not detect 21 reflectors from the 1980 inspection. The finding of the 6 new reflectors was associated with improved capabilities of the 1993 UT technique to detect and size indications above baseline noise. The inability to find 21 previously identified reflectors was attributed to the rigid positioning and step-wise movements of the mechanized scanning device used during the 1993 examination.

Of the 46 identified reflectors, 31 were sized as (axial type) underclad cracks bounded by the dimensions 0.40- to 0.70-inch long by 0.08- to 0.25-inch deep. Since the 1993 examination technique tends to oversize cracks bounded within these dimensions, the actual cracks should be smaller than indicated. None of the cracks were open to the surface. The crack growth calculations indicate that a crack 0.25-inches deep would grow to 0.299-inches deep over the next 10-year ISI interval. Based on these calculations, the cracks would remain within the acceptance standards contained in IWB-3500 and should not affect the structural integrity or design margin of the RPV nozzles.

The 15 reflectors that were not sized were all located in nozzle 17. They were identified in 1980 as reheat cracks, and were detected during the 1993 examination with an amplitude below 20-percent distance amplitude correction (DAC). Considering the large number of cracks that were sized regardless of the percent DAC on the other seven nozzles, and considering the small variations in their length and depth, the reflectors that were not sized are expected to be within the bounded dimensions of the cracks that were sized.

The indications of underclad cracks in the RPV nozzles that were identified in 1980 could not be directly compared with the 1993 inspection findings because of the inherent differences between the two inspection technologies and the robotic application in the 1993 examination. The inspections completed during the U1C6 refueling outage (1993 examination) should be available for future reproducibility checks. TVA has committed to using the data from the 1993 examination as the baseline for comparing data with the second 10-year ISI interval for Unit 1.

The reflections detected in 1980 were evaluated as cold cracks for seven nozzles and as reheat cracks for one nozzle. Finding both types of cracks in the RPV is a paradox. Cold cracks are associated with the multiple-pass cladding process performed in the absence of sufficient preheating. Reheat cracks are associated with the single-pass cladding process followed by a post-weld heat treatment. TVA used the multiple-pass cladding process on Unit 1 RPV nozzles, which calls into question the identification of reheat cracks in one out of eight nozzles in 1980. TVA did not distinguish between cold cracks and reheat cracks in the 1993 examination.

The 1980 examinations were performed using UT techniques with greater detection abilities (sensitivity) than required by the ASME Code. The 1993 UT technique was able to demonstrate that it was as sensitive as the 1980 UT technique, providing that all detectable reflectors were sized, regardless of the percent DAC. The demonstration also showed that the percent DAC could not be correlated with the crack size. This calls into question the restriction imposed by TVA to limit the sizing of reflectors from previously identified reheat cracks to only those that measured 20-percent DAC and above. Before the 1993 examination, the NRC staff requested sizing of all cracks that were detected.

In response to staff concerns to provide assurance that adequate requirements are reflected in the ISI program, TVA committed, by letter dated February 22, 1994, to change the ISI program to include the augmented ISI examinations of the RPV nozzles as follows:

(1) The ultrasonic technique for future augmented examinations will be at least as sensitive as that used to conduct the examination during the Unit 1 Cycle 6 refueling outage.

(2) The Unit 1 Cycle 6 examination will serve as the baseline for future examinations.

(3) The augmented examination will be performed near the end of the second 10-year ISI interval for Unit 1.

(4) All detected flaws will be sized, regardless of the percent distance amplitude curve (DAC).

(5) The results of the examinations will be submitted to the NRC.

(6) The augmented examinations will not be removed from the ISI program without notifying the NRC.

3.2 Unit 2

The Unit 2 RPV nozzles were cladded using a single pass weld-cladding process with a post-weld heat-treatment. This process is capable of producing reheat cracks only. The identification of reflectors detected during the 1980 supplemental examination as reheat cracks is logical. Although reheat cracks were evaluated by the NRC before the insertion of the supplemental examination to the TS, the inclusion of the supplemental examination requirement was to provide continued assurance that an adequate margin of safety would be maintained during service. Since the examination that occurred in 1980 does not correlate well with the examination conducted in 1993 on Unit 1, the same findings are expected for Unit 2. Therefore, any meaningful monitoring of underclad cracks in Unit 2 is expected to require a new baseline.

An acceptable baseline for Unit 2 can be established by performing the supplemental examination with the same UT technique used on Unit 1 in 1993 and by sizing all detected reflectors above background noise, regardless of the percent DAC. All detected reflectors must be sized because of the observation that the percent DAC does not correlate well with crack size. Once the new baseline has been established, comparisons can be made.

In response to staff concerns to provide assurance that adequate requirements are reflected in the ISI program, TVA committed, by letter dated February 22, 1994, to change the ISI program to include the augmented ISI examinations of the RPV nozzles as follows:

(1) The volumetric examinations of the reactor pressure vessel nozzles will be performed over the same cladded nozzle areas required by the ASME Code.

(2) The ultrasonic technique for the Unit 2 Cycle 6 refueling outage and future examinations will be at least as sensitive as that used to conduct the examination during the Unit 1 Cycle 6 refueling outage.

(3) The examinations performed during the Unit 2 Cycle 6 refueling outage will serve as the baseline for future examinations.

(4) All of the detected flaws will be sized regardless of the percent DAC.

(5) The results of the examinations will be submitted to the NRC.

(6) The above commitments will not be removed from the Unit 2 ISI program without notifying the NRC.

The staff has reviewed the commitments made by TVA regarding performance of the reactor vessel nozzle inspection program and the inclusion of these requirements into the ISI program. The staff has also reviewed the method used to perform the tests and evaluate the results. Based on this evaluation, the staff has determined that removal of the surveillance requirement from the TS is satisfactory.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendments. 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. The NRC staff has determined that the amendments involve 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 (58 FR 7007). Accordingly, the amendments meet 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public provided the augmented inspections outlined above are performed.

Principal Contributor: Donald Naujock
David E. LaBarge

Dated: March 15, 1994