March 30, 1992

Docket No. 50-328

Senior Vice President, Nuclear Power
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Sir:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M80499) (TS 91-08 AND 91-11)

The Commission has issued the enclosed Amendment No. 146 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Unit 2. This amendment is in response to your application dated May 24, 1991, and supplemented on August 23, 1991.

The change replaces certain cycle-specific parameter limit values in the Technical Specifications with references to a Core Operating Limits Report, in accordance with Generic Letter 88-16. In addition, changes to the Bases sections are also incorporated.

These changes affect the Technical Specifications of Unit 2 only and are being issued to coincide with the Unit 2 Cycle 5 refueling outage. Similar changes were issued as Amendment No. 155 to the Unit 1 Technical Specifications on October 23, 1991.

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, Senior Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 146 to License No. DPR-79
2. Safety Evaluation

cc w/enclosures:
See next page

*SEE PREVIOUS CONCURRENCE

OFC	PDII4/LA	PDII-4/PM	OGC	PDII-4/D
NAME	MSanders	DLaBarge dw	MYoung*	FHebdon
DATE	3/30/92	3/30/92	1/27/92	3/30 /92

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AMENDMENT NO. 146 FOR SEQUOYAH UNIT 2 - DOCKET NO. 50-328
DATED: March 30, 1992

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 146
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 May 24, 1991 and amplified August 23, 1991, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 146, 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 in conjunction with the Core Operating Limits Report within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

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 30, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 146

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf pages* are provided to maintain document completeness.

REMOVE	INSERT
I	I
II	II
1-2	1-2
1-3	1-3
1-4	1-4
1-5	1-5
1-6	1-6
1-7	1-7
1-8	1-8
B2-1	B2-1
B2-2	B2-2
3/4 1-4	3/4 1-4
3/4 1-5	3/4 1-5
3/4 1-14	3/4 1-14
3/4 1-20	3/4 1-20
3/4 1-21	3/4 1-21
3/4 1-22	3/4 1-22
3/4 1-23	3/4 1-23
3/4 2-1	3/4 2-1
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
B3/4 1-2	B3/4 1-2
B3/4 2-1	B3/4 2-1
B3/4 2-2	B3/4 2-2
B3/4 2-4	B3/4 2-4
6-22	6-22
--	6-22a

DEFINITIONS

DEFINITIONS

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.14. Unit operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

GASEOUS RADWASTE TREATMENT SYSTEM

1.15 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

MEMBERS OF THE PUBLIC

1.17 MEMBERS OF THE PUBLIC shall include all individuals who are not occupationally associated with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category does not include non-employees such a vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.5 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, or component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 The PROCESS CONTROL PROGRAM shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71; State regulations; and other requirements governing the disposal of solid radioactive wastes.

PURGE - PURGING

1.24 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHIELD BUILDING INTEGRITY

1.29 SHIELD BUILDING INTEGRITY shall exist when:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

b. The emergency gas treatment system is OPERABLE.

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.31 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee (see figure 5.1-1).

SOLIDIFICATION

1.32 Deleted.

SOURCE CHECK

1.33 Deleted.

STAGGERED TEST BASIS

1.34 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.35 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.36 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.37 An UNRESTRICTED AREA shall be any area, at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.38 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.39 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation and the W-3 correlation for conditions outside the range of WRB-1 correlation. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 or W-3 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, specified in the Core Operating Limit Report (COLR) and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}[1+ PF_{\Delta H}\ (1-P)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

$F^{RTP}_{\Delta H}$ = the $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and

$PF_{\Delta H}$ = the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR.

2.1.1 REACTOR CORE (Continued)

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature Delta T trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.1 1967 Edition, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less than 0 delta K/K/° F.

APPLICABILITY: Beginning of Cycle Life (BOL) Limit - Modes 1 and 2* only#
End of Cycle Life (EOL) Limit - Modes 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR operation in Modes 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

   2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

   3. In lieu of any other report required by Specification 6.6.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit specified in the COLR be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0
#See Special Test Exception 3.10.3

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 PPM surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 PPM surveillance limit specified in the COLR, the MTC shall be remeasured and compared to the EOL MTC limit specified in the COLR at least once per 14 EFPD during the remainder of the fuel cycle.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: Modes 1* and 2*.

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The remainder of the rods in the group with the inoperable rod are aligned to within $\pm$ 12 steps of the inoperable rod while maintaining the rod sequence and insertion limit of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

*See Special Test Exceptions 3.10.2 and 3.10.3.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the COLR:

APPLICABILITY: Modes 1* and 2*#.

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Restore the rod to within the insertion limit specified in the COLR, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.0

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the COLR.

APPLICABILITY: Modes 1* and 2*#.

ACTION:

With the control banks inserted beyond the insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the insertion limits specified in the COLR, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.0.

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the COLR;

    1. Either restore the indicated AFD to within the limits within 15 minutes, or

    2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

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3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(z)$ shall be limited by the following relationships:

$$F_Q(z) \leq \frac{[F_Q^{RTP}]\,[K(z)]}{P} \text{ for } P > 0.5$$

$$F_Q(z) \leq \frac{[F_Q^{RTP}]\,[K(z)]}{0.5} \text{ for } P \leq 0.5$$

where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the COLR,

$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

$K(z)$ = the normalized $F_Q(z)$ as a function of core height specified in the COLR.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; THERMAL POWER may then be increased provided $F_Q(z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.2 $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3 percent to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{P \times W(z)} \quad \text{for } P > 0.5$$

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{W(z) \times 0.5} \quad \text{for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z), and W(z) are specified in the COLR as per Specification 6.9.1.14.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10 percent or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 effective full power days, whichever occurs first.

*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M (z)}{K(z)} \right]$$

has increased since the previous determinatin of $F_Q{}^M(z)$ either of the following actions shall be taken:

1. $F_Q{}^M(z)$ shall be increased by 2 percent over that specified in 4.2.2.2.c, or

2. $F_Q{}^M(z)$ shall be measured at least once per 7 effective full power days until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M (z)}{K(z)} \right] \text{ is not increasing.}$$

f. With the relationships specified in 4.2.2.2.c above not being satisfied:

1. Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q{}^M(z) \times W(z)}{\frac{F_Q^{RTP}}{P} \times K(z)} \right] \right) -1 \right\} \times 100 \qquad \text{for } P \geq 0.5$$

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q{}^M(z) \times W(z)}{\frac{F_Q^{RTP}}{0.5} \times K(z)} \right] \right) -1 \right\} \times 100 \qquad \text{for } P < 0.5$$

2. Either of the following actions shall be taken:

   a. Place the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied. Power level may then be increased provided the AFD limits of Specification 3 are reduced 1% AFD for each percent $F_Q(z)$ exceeded its limit, or

   b. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above.

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3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The Nuclear Enthalpy Hot Channel Factor, $F_{\Delta H}^{N}$ shall be limited by the following relationship:

$$F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP} [1.0 + PF_{\Delta H} (1.0-P)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

$F_{\Delta H}^{RTP}$ = The $F_{\Delta H}^{N}$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and

$PF_{\Delta H}$ = The power factor multiplier for $F_{\Delta H}^{N}$ specified in the COLR.

APPLICABILITY: MODE 1

ACTION:

With $F_{\Delta H}^{N}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≦ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate thru in-core mapping that $F_{\Delta H}^{N}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F_{\Delta H}^{N}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER; at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting end of cycle life (EOL) MTC value. The 300 PPM surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting EOL MTC value.

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within it analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in a OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two separate and redundant boron injection system are provided to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F^N_{WH} Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(z)$ upper bound envelope of the F_Q limit specified in the COLR times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The compuer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the allowed WI-Power operating space and the THERMAL POWER is greater than 50 percent of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS

The limits on heat flux hot channel factor and nuclear enthalpy hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 13 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The $F^N_{\Delta H}$ limit as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. The 5% is the appropriate allowance for a full core map taken with the in-core detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the in-core detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq F^{RTP}_{\Delta H}/1.08$. The 8% allowance is based on the following considerations.

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q.

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics test can be compensated for in F_Q by restricting axial flux distribution. This compensation for $F^N_{\Delta H}$ is less readily available.

Fuel rod bowing reduces the value of DNB ratio. Margin has been retained between the DNBR value used in the safety analysis (1.38) and the WRB-1 correlation limit (1.17) to completely offset the rod bow penalty.

The applicable value of rod bow penalty is referenced in the FSAR.

Margin in excess of the rod bow penalty is available for plant design flexibility.

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor W(z), to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(z) function is specified in the COLR.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the safety analysis DNBR limit throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or Safety Valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.14 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference Limits for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, K(z), and W(z) for Specification 3/4.2.2, and
6. Nuclear Enthalpy Hot Channel Factor and Power Factor Multiplier for Specification 3/4.2.3.

6.9.1.14.a The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Hot Channel Factor.)
2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION", JUNE 1983 (W Proprietary).
   (Methodology for Specification 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(z) surveillance requirements for F_Q Methodology).)
3. WCAP-10266-P-A Rev. 2, "THE 1981 REVISION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

6.9.1.14.b The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

CORE OPERATING LIMITS REPORT (Continued)

6.9.1.14.c THE CORE OPERATING LIMITS REPORT shall be provided within 30 days after cycle start-up (Mode 2) for each reload cycle or within 30 days of issuance of any midcycle revision to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2.1 Special reports shall be submitted within the time period specified for each report, in accordance with 10 CFR 50.4.

6.9.2.2 Diesel Generator Reliability Improvement Program

As a minimum the Reliability Improvement Program report for NRC audit, required by LCO 3.8.1.1, Table 4.8-1, shall include:

(a) a summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed

(b) analysis of failures and determination of root causes of failures

(c) evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the Plant

(d) identification of all actions taken or to be taken to 1) correct the root causes of failures defined in b) above and 2) achieve a general improvement of diesel generator reliability

(e) the schedule for implementation of each action from d) above

(f) an assessment of the existing reliability of electric power to engineered-safety-feature equipment

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

ENCLOSURE 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 146 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-328

1.0 INTRODUCTION

By letter dated May 24, 1991, and supplemented by letter dated August 23, 1991, the Tennessee Valley Authority (the licensee) proposed changes to the Technical Specifications (TS) for the Sequoyah Nuclear Plant, Units 1 and 2. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of the TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988.

The licensee's letter dated August 23, 1991, provided clarifying information and changes to the TS Bases that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definition section of the TS was modified to include a definition of the COLR that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with Nuclear Regulatory Commission (NRC) approved methodologies that maintain the limits of the safety analysis. The definition indicates that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits:

(a) Specification 3.1.1.3 and Surveillance Requirement 4.1.1.3

The moderator temperature coefficient (MTC) limits for this specification and for this surveillance requirement are specified in the COLR.

(b) Specification 3.1.3.5 and Surveillance Requirement 4.1.3.5

The shutdown bank insertion limit for this specification and for this surveillance requirement is specified in the COLR.

(c) Specification 3.1.3.6

The control bank insertion limits for this specification are specified in the COLR.

(d) Specification 3.2.1

The axial flux difference limits as a function of rated thermal power for this specification are specified in the COLR.

(e) Specification 3.2.2 and Surveillance Requirement 4.2.2

The total peaking factor (F_Q) limit at rated thermal power, the normalized F_Q limit as a function of core height K(z), and the cycle dependent function that accounts for power distribution transients encountered during normal operation, W(z), for this specification and for this surveillance requirement are specified in the COLR.

(f) Specification 3.2.3

The nuclear enthalpy rise hot channel factor (F^N-delta-H) limit at rated thermal power and the power factor multiplier (PF-delta-H) for this specification are specified in the COLR.

Changes to the bases of the affected specifications and Basis 2.1.1 have been provided by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

(3) Specification 6.9.1.14 is revised to delete a previous reporting requirement on Peaking Factor Limit Report and to add the COLR to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC approved methodologies and be consistent with all applicable limits of the safety analysis. The approved Westinghouse (W) methodologies are the following:

(a) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limit, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

(b) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification," June 1983 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor W(z) surveillance requirements for F_Q methodology)).

(c) WCAP-10266-P-A, Revision 2, "The 1981 Version of Westinghouse Evaluation Model Using BASH Code," March 1987 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to Generic Letter 88-16 for removing cycle-specific parameter limits from the TS. Because plant operation continues to be limited in accordance with the values of the cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

The licensee also proposed a change to the Action Statement c.2 of Specification 3.1.3.1, to reference Specification 3.1.3.6 instead of Figure 3.1-1. This change was necessary because the figure has been relocated to the COLR. Consequently, this change is administrative in nature and is acceptable. Other changes to the Bases sections proposed by the licensee consist of information which clarifies the referenced sections. They are, therefore, acceptable.

3.0 SUMMARY

We have reviewed the request by the Tennessee Valley Authority to modify the Technical Specifications of the Sequoyah plants that would remove the specific values of some cycle-dependent parameters from the specifications and place the values in a Core Operating Limits Report that would be referenced by the specifications. Based on this review, we conclude that these Technical Specification modifications are acceptable.

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 amendment changes 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 to the Surveillance Requirements. In addition, this amendment changes reporting or administrative procedures or requirements. The NRC staff has determined that the amendment involves no significant increase the the amounts, and no signficant 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 31443). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: T. Huang

Date: March 30, 1992