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

May 8, 1990

Docket No. 50-327

Mr. Oliver D. Kingsley, Jr.
Senior Vice President, Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley:

SUBJECT: VANTAGE 5 HYBRID FUEL (TAC 75751) (TS 89-33) - SEQUOYAH NUCLEAR PLANT, UNIT 1

The Commission has issued the enclosed Amendment No. 138 to Facility Operating License No. DPR-77 for the Sequoyah Nuclear Plant, Unit 1. This amendment is in response to your application dated January 12, 1990 and the supplemental letter dated April 13, 1990.

This amendment modifies the Sequoyah Nuclear Plant, Unit 1, Technical Specifications (TSs) to permit the use of the Vantage 5 Hybrid fuel at Unit 1 in the upcoming Operating Cycle 5. The unit is currently in the Cycle 4 refueling outage and is scheduled to restart from this outage to begin the Operating Cycle 5 in late May 1990.

The changes to the TSs are the following: (1) revise the TS Bases for safety limits to refer to the WRB-1 correlation and to the associated safety analysis limit for the departure from nucleate boiling ratio (DNBR); (2) revise TS 3.1.3.4 to incorporate a new rod drop time of less than or equal to 2.7 seconds; (3) revise TS 3.2.3 to delete the rod bow penalty as a function of burnup in the Nuclear Enthalop Hot Channel Factor equation and delete Figures 3.2-3 and 3.2-4; (4) revise Table 3.2-1 and TS 3.2.5 to define the reactor coolant system (RCS) total flow rate limit, including uncertainties, to be 378,400 gallons per minute (gpm) for the departure from nucleate boiling parameters and add surveillance requirements for the RCS total flow rate; and (5) revise the Bases for TSs 3.2.3, 3.2.5, and 3.4.1. The title in the index for Section 3/4.2.3 and Bases Sections 3.4.2.2 and 3/4.2.3 of the TSs are also revised.

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

Sincerely,

JnDonohew for
Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 138 to License No. DPR-77
2. Safety Evaluation

cc w/enclosures:
See next page

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AMENDMENT NO. 138 FOR SEQUOYAH UNIT NO. 1 - DOCKET NO. 50-327 and
DATED: May 8, 1990

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.138
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 12, 1990 and the supplemental letter dated April 13, 1990 comply 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.

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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. 138 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Jackie Donohew Jr for
Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NO. 138

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

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
V	V
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XII	XII
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B 2-2	B 2-2*
B 2-3	B 2-3
B 2-4	B 2-4*
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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, $F^N_{\Delta H}$, of 1.55 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:

$$F^N_{\Delta H} = 1.55\ [1 + 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

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.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the safety analysis DNBR limit for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for all single or multiple dropped rods.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate

Range Channels will initiate a reactor trip at approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Delta T

The Overtemperature Delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature Delta T setpoint. Three loop operation above the 4 loop P-8 setpoint is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature Delta T channels and raising the P-8 setpoint to its 3 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower Delta T

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident

analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 89% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 89% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the safety analysis DNBR limit during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature Delta T trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature Delta T trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below the safety analysis DNBR limit during normal operational transients and anticipated transients with 3 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 1.5×10^6 lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 24 percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.6 seconds.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position# shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 71% of RATED THERMAL POWER

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

#Fully withdrawn shall be the condition where shutdown and control banks are at a position within the interval of $\geq$ 222 and $\leq$ 231 steps withdrawn, inclusive.

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

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

Where:

a. $F^N_{\Delta H} \leq 1.55\ [1.0 + 0.3\ (1.0-P)]$

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

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ 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 $\leq$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ 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% of greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The measured $F^N_{\Delta H}$ shall be increased by 4% for measurement uncertainty.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until:

a) Either the QUADRANT TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours:

a) Either reduce the QUADRANT POWER TILT RATIO to within its limit, or

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

ACTION: (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until:

a) Either the QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until:

a) Either the QUADRANT POWER TILT RADIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

ACTION: (Continued)

2. 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 less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. With the indicated QUADRANT POWER TILT RATIO not confirmed as required by Surveillance Requirement 4.2.4.2, reduce THERMAL POWER to less than 75 percent RATED THERMAL POWER within 6 hours.

e. With the QUADRANT POWER TILT RATIO not monitored as required by Surveillance Requirement 4.2.4.1, reduce THERMAL POWER to less than 50 percent of RATED THERMAL POWER within the next 6 hours.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE.

b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 percent of RATED THERMAL POWER with one Power Range Channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from the 4 pairs of symmetric thimble locations or from performance of a full core map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System (RCS) T_{avg}.

b. Pressurizer Pressure

c. RCS Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The RCS total flow rate shall be determined by measurement at least once per 18 months.

4.2.5.3 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 583°F
Pressurizer Pressure	$\geq$ 2220 psia*
Reactor Coolant System Total Flow	$\geq$ 378400 gpm#

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER, physics test, or performance of surveillance requirement 4.1.1.3.b.

Includes a 3.5% flow measurement uncertainty.

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.

$F^N_{\Delta H}$ 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 assure that the $F_Q(Z)$ upper bound envelope of 2.32 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 computer 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 ΔI-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 the heat flux hot channel factor and the nuclear enthalpy rise 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 ± 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 $F^N_{\Delta H}$ 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 incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When an $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore 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 $F^N_{\Delta H} \leq$ 1.55/1.08. The 8% allowance is based on the following considerations.

a. abnormal perturbatios 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 or 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 for normal operation is provided in the Peaking Factor Limit Report per Specification 6.9.1.14.

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.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above the safety analysis DNBR limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, a single reactor coolant loop or residual heat removal (RHR) loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

In MODE 5, single failure considerations require that two RHR loops be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

3/4.4.2 and 3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no

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

ENCLOSURE 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 138 TO FACILITY OPERATING LICENSE NO. DPR-77

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-327

1.0 INTRODUCTION

By letter dated January 12, 1990, the Tennessee Valley Authority (TVA or the licensee) submitted a request for changes to the Sequoyah Nuclear Plant, Units 1 and 2, Technical Specifications (Tss) which are needed for the use of VANTAGE 5 Hybrid (V5H) fuel assemblies for the Sequoyah, Units 1 and 2 reload core and future cores. The V5H fuel design evolved from the V5H Optimized Fuel Assembly (OFA) and Standard (STD) fuel assembly designs. The features of the V5H fuel assembly consist of reconstitutable or removable top nozzles, integral fuel burnable adsorbers, lower-pressure drop and snag-resistant grids, debris filter bottom nozzles (DFBNs), and the capability of achieving extended burnups. These features were previously reviewed and approved by NRC in the Westinghouse Electric Corporation topical report WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly," Addendum 2 (REF 1).

In its letter, the licensee stated that the evaluations performed for this new fuel accomodate the effects from the following modifications that are planned for the Cycle 4 refueling outage in 1990 for each unit:

1. Resistance temperature detector bypass elimination,

2. Eagle 21 digital protection system,

3. Upper head injection removal,

4. Boron injection tank deactivation,

5. New steamline break protection, and

6. Reactor trip on steam flow/feed flow mismatch.

Unit 1 is currently in its Cycle 4 refueling outage and has put the new fuel in the core. The new fuel will be used in the Unit 1 Operating Cycle 5 once it has restarted from the refueling outage. Unit 2 will shutdown for its Cycle 4 refueling outage in the fall of 1990. The new fuel will be put in the core at that time.

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Therefore, as a result of this fuel upgrade for Unit 1, TVA proposed to modify the Unit 1 TSs for the following changes: (1) revise the TSs Bases for safety limits to refer to the WRB-1 correlation and to the associated safety analysis limit for departure from nucleate boiling ratio (DNBR); (2) revise TS 3.1.3.4 to incorporate a new rod drop time of less than or equal to 2.7 seconds; (3) revise TS 3.2.3 to delete the rod bow penalty as a function of burnup in the FNH (Nuclear Enthalpy Hot Channel Factor) equation and delete Figures 3.2-3 and 3.2-4; (4) revise Table 3.2-1 and TS 3.2.5 to define the reactor coolant system (RCS) total flow rate limit, including uncertainties, to be 378,400 gallons per minute (gpm) for the departure from nucleate boiling (DNB) parameters and add surveillance requirements for RCS total flow rate; and (5) revise the Bases for TSs 3.2.3, 3.2.5, and 3.4.1. The titles in the index of the TSs for the following sections would also be revised: Section 3/4.2.3, Bases Section 3/4.2.2 and Bases Section 3/4.2.3.

The licensee provided additional information concerning the fuel criteria used for the locked rotor analysis for the V5H fuel in its letter dated April 13, 1990. This information does not change the substance of the proposed action published in the Federal Register Notice (55 FR 4281) on February 7, 1990 and does not affect the staff's initial determination of no significant hazards consideration in that notice.

2.0 EVALUATION

During the staff's review of the VANTAGE 5 fuel design in WCAP-10444-P-A, the staff identified a few conditions to be resolved for licensees who proposed using the VANTAGE 5 fuel design. Since the V5H fuel design adopts some features from the VANTAGE 5 fuel design, the staff's evaluation will address those conditions listed in the Safety Evaluation (SER) for WCAP-10444-P-A which could affect Sequoyah's V5H fuel. These conditions are the following:

(1) Statistical Convolution Method

In the SER on WCAP-10444, the staff stated that the statistical convolution method should not be used in the VANTAGE 5 fuel for evaluating the fuel rod shoulder gap. The licensee stated that the statistical convolution method was not used for the V5H fuel design for Sequoyah and the currently NRC approved method was used for evaluating the fuel rod shoulder gap. Therefore, the staff concludes that this is acceptable and the licensee has met this condition.

(2) Irradiation Demonstration Program

In the SER on WCAP-10444, the staff required that an irradiation program be performed to confirm the VANTAGE 5 fuel performance. The licensee stated that there were numerous demonstration programs involving OFA fuel assemblies containing Zircaloy grids irradiated in 14x14, 15x15, and 17x17 fuel assembly array cores. Sequoyah has a 17x17 core. The satisfactory performance of these demonstration assemblies resulted in OFA with Zircaloy grids reload applications in many Westinghouse reactors. The OFA fuel assemblies with Zircaloy grids cover the V5H fuel design features for Sequoyah; therefore, the staff concludes that the V5H fuel assemblies should perform satisfactorily in Sequoyah and the licensees has met this condition.

(3) Improved Thermal Design Procedure (ITDP)

In the SER on WCAP-10444, the staff stated that those restrictions in approving the use of Westinghouse improved thermal design procedure (ITDP) should be applied to the VANTAGE 5 fuel design. The licensee stated that they conform to these restrictions of ITDP for Sequoyah. The staff, therefore, concludes that this is acceptable and the licensee has met this condition.

(4) DNBR Limit

In the SER on WCAP-10444, the staff stated that a plant-specific analysis should be performed to show that the DNBR limit is not violated with the higher value of FNH. The licensee stated that the V5H fuel for Sequoyah does not employ higher values of FNH, thus no reanalysis of DNBR transients is needed. The staff, therefore, concludes that this condition is satisfied for V5H fuel at Sequoyah and the licensee has met this condition.

(5) Positive Moderator Temperature Coefficient (MTC)

In the SER on WCAP-10444, the staff stated that if a positive moderator temperature coefficient (MTC) is intended, the same positive MTC should be used in the plant-specific analysis. The licensee stated that no positive MTC was considered for Sequoyah. The staff, therefore, concludes that this condition is satisfied for V5H fuel at Sequoyah.

(6) Reactor Coolant Pump Shaft Seizure

In the SER on WCAP-1044, the staff stated that the mechanistic approach (2700°F peak clad temperature) in determining the fraction of fuel failures during the reactor coolant pump seizure accident was unacceptable and the fuel failure criterion should be the 95/95 DNBR limit. The licensee reanalyzed the reactor coolant pump shaft (locked rotor) accident based on a failure criterion of 95/95 DNBR limit for V5H fuel. The licensee concluded that the fuel rod failure rate is less than 10% of the total rods in the core, which is bounded in the FSAR analysis. The staff, therefore, concludes that the reactor coolant pump shaft seizure accident is adequately addressed for V5H fuel at Sequoyah.

2.2 Technical Specification Changes

The proposed TS changes are related to the use of the V5H fuel at Sequoyah, a new DNBR correlation, and a new rod bow penalty methodology. The proposed changes are evaluated below:

(1) TS Bases Section 2.1.1, Pages B2-1, B2-3, and B2-5 for Units 1 and 2

The old W-3 DNBR correlation is changed to the WRB-1 correlation for standard and V5H fuel designs. This DNBR correlation has been approved by the staff for use in licensing applications. The staff concludes that these changes are acceptable.

(2) TS Bases Section 3/4.2.5

The phrase "a minimum DNBR of 1.30" is changed to "a minimum DNBR greater than or equal to the safety analysis DNBR limit" because there is only one DNBR correlation intended for two different fuel designs. The staff concludes that this change is acceptable.

(3) TS Section 3.1.3.4

The rod drop time is revised to be less than or equal to 2.7 seconds due to the use of the V5H fuel. The increased rod drop time is due to the reduced guide tube diameter for the V5H fuel grids and the resulting increased dash pot effect. The licensee has taken into account the effect of the increased rod drop time in the safety analyses. The staff concludes that this change is acceptable.

(4) TS Section 3/4.2.3, and TS Bases Sections 3/4.2.2 and 3/4.2.3

The rod bow penalty is revised to incorporate a new methodology which reduces the rod bow penalty. The Figures 3.2-3 and 3.2-4 are being deleted. New statistical methods have been developed by Westinghouse that verify that the past treatment of rod bow penalty provided an overestimation of the affects on DNB. Application of the new methods to Sequoyah for the standard and the V5H fuel has verified the reduction in rod bow penalty. The reduction allows for accomodation of the entire penalty in the establishment of the DNBR safety limit. The licensee has demonstrated that the use of new DNBR correlations has enough margin to offset the rod bow penalty at burnups greater than 24,000 MWd/MTU. The staff concludes that the proposed reduced rod bow penalty is acceptable for Unit 1.

(6) TS Section 3/4.2.5

These proposed changes are to include the reactor coolant system (RCS) total flow rate in the list of DNB parameters. The RCS flow limit and its associated surveillance requirements have been moved from TS 3/4.2.3 to TS 3/4.2.5, DNB parameters, which now establishes a minimum allowable RCS flow to prevent violation of the DNB safety limit during normal operation and accident conditions. The maximum limit of 378,400 gpm was established for this flow. The staff concludes that these change are acceptable for Units 1 and 2.

2.3 Conclusion

The staff has evaluated the licensee's submittal on the V5H fuel design and the proposed TS changes for Sequoyah Units 1 and 2 transition cores and all future V5H cores. Based on the NRC approved generic topical report WCAP-10444-P-A and the plant-specific Sequoyah analyses discussed above, the staff concludes that the use of V5H fuel at Sequoyah and the proposed TS changes for Sequoyah Units 1 and 2 are acceptable.

The transition cores are the cores starting from the reload core for the Operating Cycle 5 until all the fuel in the core are the V5H fuel. Unit 1 is in its Cycle 4 refueling outage preparing to restart from the outage and begin its Operating Cycle 5. Therefore, the Unit 1 TSs will be changed at this time. The Unit 2 TSs will be changed during its Cycle 4 refueling outage which is scheduled to begin in the fall of 1990.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the 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. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 4281) on February 7, 1990, and consulted with the State of Tennessee. No public comments were received and the State of Tennessee did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

5.0 REFERENCE

1.0 Letter from C. Thomas (NRC) to E.P. Rahe (Westinghouse Electric Corporation), Subject: "Acceptance For Referencing of Licensing Topical Report WCAP-10444, VANTAGE 5 Fuel Assembly;" dated July 18, 1985.

Principal Contributor: S. Wu

Dated: May 8, 1990