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Docket No. 50-327

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JUN 26 1981

Docket No. 50-327

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

SUBJECT: ISSUANCE OF AMENDMENT NO. 7 TO FACILITY OPERATING LICENSE NO. DPR-77 - SEQUOYAH NUCLEAR PLANT, UNIT 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 7 to Facility Operating License No. DPR-77.

This amendment changes the Technical Specifications of Unit 1 to allow deletion of the reactor trip on turbine trip below 50-percent power level. These anticipatory trip modifications were proposed in your letters dated May 13 and 14, 1981.

A copy of the related safety evaluation supporting Amendment No. 7 to Facility Operating License DPR-77 is enclosed. Also enclosed is a copy of the Federal Register Notice which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/S/

Elinor G. Adensam, Acting Chief
Licensing Branch #4
Division of Licensing

cp

Enclosures:
1. Amendment No. 7
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

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JUN 30 1981
U.S. NUCLEAR REGULATORY COMMISSION

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PDR ADOCK 05000327
P PDR

OFFICE	DL:LB #4	LA:LB #4/DL	OELD	DL:LB #4			
SURNAME	TKenyon/hmc	MService	McCormick	EAdensam			
DATE	6/ /81	6/ /81	6/25/81	6/ /81			

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Resident Inspector/Sequoyah NPS
c/o U.S. Nuclear Regulatory Commission
P. O. Box 699
Hixson, Tennessee 37343

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. DPR-77

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

   A. The application for amendment to the Sequoyah Nuclear Plant, Unit 1 (the facility) Facility Operating License No. DPR-77 filed by the Tennessee Valley Authority (licensee), dated May 13 and 14, 1981 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assuance (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 amended license is hereby amended by page changes to the Appendix A Technical Specifications as indicated in the attachments to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-077 is hereby amended to read as follows:

OFFICE	8107070516 810626					
SURNAME	PDR ADOCK 05000327					
DATE	P PDR					

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 7, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This amended license is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Elinor G. Adensam, Acting Chief
Licensing Branch #4
Division of Licensing

Attachment:
Appendix A Technical
Specification Changes

Date of Issuance: June 26, 1981

concurrence in form only

OFFICE	DL:LB #4	LA:DL:LB#4	OELD	DL:LB #4			
SURNAME	TJKenyon/hmc	MService	McGurren	EAdensam			
DATE	6/[illegible]/81	6/[illegible]/81	6/25/81	6/[illegible]/81			

ATTACHMENT TO LICENSE AMENDMENT NO. 7

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contains vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Overleaf Page	Amended Page
2-8	2-7
3/4 3-3	B2-7
3/4 3-8	3/4 3-4
	3/4 3-7

OFFICE							
SURNAME							
DATE							

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
21. Turbine Impulse Chamber Pressure - (P-13) Input to Low Power Reactor Trips Block P-7	≤ 10% Turbine Impulse Pressure Equivalent	≤ 11% Turbine Impulse Pressure Equivalent
22. Power Range Neutron Flux - (P-8) Low Reactor Coolant Loop Flow, and Reactor Coolant Pump Breaker Position	≤ 35% of RATED THERMAL POWER	≤ 36% of RATED THERMAL POWER
23. Power Range Neutron Flux - (P-10) - Enable block of Source, Intermediate, and Power Range (low setpoint) reactor Trips	≥ 10% of RATED THERMAL POWER	≥ 9% of RATED THERMAL POWER
24. Reactor Trip P-4	Not Applicable	Not Applicable
25. Power Range Neutron Flux - (P-9) - Blocks Reactor Trip for Turbine Trip Below 50% Rated Power	≤ 50% of RATED THERMAL POWER	≤ 51% of RATED THERMAL POWER

NOTATION

NOTE 1: Overtemperature $\Delta T \left(\frac{1}{1 + \tau_1 S}\right) \leq \Delta T_o \{K_1 - K_2 \left(\frac{1 + \tau_2 S}{1 + \tau_3 S}\right)[T\left(\frac{1}{1 + \tau_4 S}\right) - T'] + K_3(P - P') - f_1(\Delta I)\}$

where: $\frac{1}{1 + \tau_1}$ = Lag compensator on measured ΔT

τ_1 = Time constants utilized in the lag compensator for $\Delta T_3 \tau_1 = 2$ secs.

ΔT_o = Indicated ΔT at RATED THERMAL POWER

$K_1 \leq 1.14$

$K_2 = 0.009$

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

$\frac{1 + \tau_2 S}{1 + \tau_3 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_2, & τ_3 = Time constants utilized in the lead-lag controller for T_{avg}, τ_2 = 33 secs., τ_3 = 4 secs.

T = Average temperature °F

$\frac{1}{1 + \tau_4 S}$ = Lag compensator on measured T_{avg}

τ_4 = Time constant utilized in the measured T_{avg} lag compensator, τ_4= 2 secs.

T' $\leq$ 578.2°F (Nominal T_{avg} at RATED THERMAL POWER)

K_3 = 0.00043

P = Pressurizer pressure, psig

P' = 2235 psig (Nominal RCS operating pressure)

S = Laplace transform operator (sec^{-1})

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 30 percent and + 4 percent $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

BASES

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-7. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions on increasing power:

P-6 Enables the manual block of the source range reactor trip (i.e., prevents premature block of source range trip).

P-7
P-13 Defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump undervoltage and underfrequency, pressurizer low pressure, and pressurizer high level.

P-8 Defeats the automatic block of reactor trip on low RCS coolant flow in a single loop.

P-9 Defeats the automatic block of Reactor Trip on Turbine Trip.

P-10 Enables the manual block of reactor trip on power range (low setpoint), intermediate range, as a backup block for source range, and intermediate range rod stops (i.e., prevents premature block of the noted functions).

On decreasing power, the opposite function is performed at reset setpoints.

P-4 Reactor tripped - Actuates turbine trip, closes main feedwater valves on T_{avg} below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows manual block of the automatic reactuation of safety injection.

Reactor not tripped - defeats manual block preventing automatic reactuation of safety injection.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	7#
14. Main Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2	7#
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in same loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	7#
16. Undervoltage-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
17. Underfrequency-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
18. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	7#
B. Turbine Stop Valve Closure	4	4	4	1	13

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
19. Safety Injection Input from ESF	2	1	2	1, 2	1
20. Reactor Trip Breakers	2	1	2	1, 2, and *	1
21. Automatic Trip Logic	2	1	2	1, 2, and *	1
22. Reactor Trip System Interlocks					
A. Intermediate Range Neutron Flux P-6	2	1	2	2, and*	8a
B. Power Range Neutron Flux - P-7	4	2	3	1	8b
C. Turbine Impulse Chamber Pressure - P-13	2	1	2	1	8b
D. Power Range Neutron Flux - P-8	4	2	3	1	8c
E. Power Range Neutron Flux - P-10	4	2	3	1, 2	8d
F. Power Range Neutron Flux - P-9	4	2	3	1	8e
G. Reactor Trip - P-4	2	1	2	1, 2, and*	14

TABLE 3.3-1 (Continued)

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, declare the interlock and all affected channels of the functions listed below inoperable and apply the appropriate ACTION statement(s). Functions to be evaluated are:

a. Source Range Reactor Trip.

b. Reactor Trip

Low Reactor Coolant Loop Flow (2 loops)
Undervoltage
Underfrequency
Pressurizer Low Pressure
Pressurizer High Level

c. Reactor Trip

Low Reactor Coolant Loop Flow (1 loop)

d. Reactor Trip

Intermediate Range
Low Power Range
Source Range

e. Reactor Trip

Turbine Trip

ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

ACTION 10 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below the P-8 (Block Low Reactor Coolant Pump Flow) setpoint breaker within the next 2 hours. Operation below the P-8 (Block of Low Reactor Coolant Pump Flow) setpoint breaker may continue pursuant to ACTION 11.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

TABLE 3.3-1 (Continued)

ACTION 13 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level above the P-7 (Block of Low Power Reactor Trips) setpoint, place the inoperable channel in the tripped condition within 1 hour, operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 14 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

SAFETY EVALUATION REPORT BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 7

TO FACILITY OPERATING LICENSE DPR-77

TENNESSEE VALLEY AUTHORITY

NUREG - 0737 Requirement (Item II.K.3, C.3.10)

For Westinghouse-designed reactors, if the anticipatory reactor trip upon turbine trip is to be modified to be bypassed at power levels less than 50 percent, rather than below 10 percent as in current design, demonstrate that the probability of a small break LOCA resulting from a stuck open PORV is not significantly changed by this modification.

Discussion

Analyses performed for the Sequoyah Units have shown that a 26 psi margin between peak system transient pressure and the PORV actuation setpoint is predicted for turbine trip without anticipatory reactor trip at 50 percent of rated power.

The LOFTRAN computer code used to perform these predictions for Sequoyah Units 1 and 2 is under review by the staff and has been used by Westinghouse in previous comparisons (one of these being Farley-Unit 2) with plant startup tests for 50 percent load rejection where code predictions were found to be conservative relative to test results. Neither the tests nor the predictions have resulted in, or shown the occurrence of, PORV challenges.

The predictions computed for the Sequoyah units are further supported by the most recent results measured in a turbine trip test performed at 50% power on the Farley-Unit 2 plant, which is similar in licensed power level and design to the Sequoyah units.

This Farley test, performed to verify analytical predictions, demonstrated a peak transient pressurizer pressure of 2260 psia, allowing a 90 psi margin to the PORV setpoint pressure. The predicted margin had been 50 psi, so that the predicted peak pressure was shown to be conservative.

Based on the computed predictions for Sequoyah Units 1 and 2, and the tests results obtained from another similar licensed plant, it is concluded that operation of the plant at 50 percent rated power or below without the anticipatory reactor trip tie to the turbine trip will not significantly change the probability of a small break LOCA due to a stuck open PORV, thereby satisfying the requirements of this item of the Action Plan.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the

the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 26, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-327

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT

FACILITY OPERATING LICENSE NO. DPR-77

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 7 to Facility Operating License No. DPR-77, issued to Tennessee Valley Authority (licensee) for the Sequoyah Nuclear Plant, Unit 1 (the facility) located in Hamilton County, Tennessee. This amendment changes the Technical Specifications to allow deletion of the reactor trip on turbine trip below 50-percent power level.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5 (d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

OFFICE	8107070521 810626						
SURNAME	PDR ADOCK 05000327 P PDR						
DATE							

For further details with respect to this action, see (1) Tennessee Valley Authority letters dated May 13 and 14, 1981, (2) Amendment No. 7 to Facility Operating License No. DPR-77 with Appendix A Technical Specification page changes, and (3) the Commission's related Safety Evaluation.

All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C., and the Chattanooga Hamilton County Bicentennial Library, 1001 Broad Street, Chattanooga, Tennessee 37402. A copy of Amendment No. 7 may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 26th day of June, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam, Acting Chief
Licensing Branch No. 4
Division of Licensing

concurrence on notice only

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DATE	6/[illegible]/81	6/[illegible]/81	6/25/81	6/[illegible]/81			