May 12, 1987

Docket Nos.: 50-327
and 50-328

Mr. S. A. White
Manager of Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. White:

Subject: Issuance of Amendment No. 55 to Facility Operating License No. DPR-77 and Amendment No. 47 to Facility Operating License No. DPR-79 - Sequoyah Nuclear Plant, Units 1 and 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No.55 to Facility Operating License No. DPR-77 and Amendment No. 47 to Facility Operating License No. DPR-79. These amendments are in response to your request dated December 23, 1986.

The amendments change the Technical Specifications to permit installation of slower acting valve operators on valves between the suction of the centrifugal charging pumps and the volume control tank. The amendments are effective as of their date of issuance. This letter should not be construed as an authorization to commence operations prior to the Tennessee Valley Authority appropriately addressing the concerns identified in the 50.54(f) letter dated September 17, 1985.

A copy of the related safety evaluation supporting Amendment No. 55 to Facility Operating License DPR-77 and Amendment No. 47 to Facility Operating License DPR-79 is enclosed.

Notice of issuance will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

Original signed by

John A. Zwolinski, Assistant Director
for Projects
Division of TVA Projects
Office of Special Projects

Enclosures:
1. Amendment No. 55 to DPR-77
2. Amendment No. 47 to DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

PWR#4/DPWR-A	PWR#4/DPWR-A	PWR#4/DPWR-A	TVAPS/~~OSP~~ SPR
MDuncan/rad	JHolonich	BJYoungblood	JZwolinski
03/3 /87	03/3 /87	03/16/87	04/29/87

8705260001 870512
PDR ADOCK 05000327
P PDR

Mr. S. A. White
Tennessee Valley Authority

Sequoyah Nuclear Plant

cc:
Tennessee Department of
Public Health
ATTN: Director, Bureau of
Environmental Health Services
Cordell Hull Building
Nashville, Tennessee 37219

Mr. R. W. Cantrell
ATTN: D.L. Williams
Tennessee Valley Authority
400 West Summit Hill Drive, W12 A12
Knoxville, Tennessee 37902

Mr. Bob Faas
Westinghouse Electric Corp.
P.O. Box 355
Pittsburgh, Pennsylvania 15230

Mr. R. L. Gridley
Tennessee Valley Authority
5N 157B Lookout Place
Chattanooga, Tennessee 37402-2801

Mr. M. R. Harding
Tennessee Valley Authority
Sequoyah Nuclear Plant
P.O. Box 2000
Soddy Daisy, Tennessee 37379

Resident Inspector/Sequoyah NP
c/o U.S. Nuclear Regulatory Commission
2600 Igou Ferry Road
Soddy Daisy, Tennessee 37379

Mr. H.L. Abercrombie
Tennessee Valley Authority
Sequoyah Nuclear Plant
P.O. Box 2000
Soddy Daisy, Tennessee 37379

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission,
101 Marietta Street, N.W.
Atlanta, Georgia 30323

Mr. Michael H. Mobley, Director
Division of Radiological Health
T.E.R.R.A. Building
150 9th Avenue North
Nashville, Tennessee 37203

County Judge
Hamilton County Courthouse
Chattanooga, Tennessee 37402

Mr. Richard King
c/o U.S. GAO
1111 North Shore Drive
Suite 225, Box 194
Knoxville, Tennessee 37919

AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. DPR-77 - Sequoyah Nuclear Plant Unit 1
AMENDMENT NO. 47 TO FACILITY OPERATING LICENSE NO. DPR-79 - Sequoyah Nuclear Plant Unit 2

DISTRIBUTION w/ enclosures:
Docket Nos. 50-327/328
NRC PDR
Local PDR
Projects Reading File
J. Axelrad
S. Ebneter
S. Richardson
B. D. Liaw
S. R. Connelly, OIA
J. Holonich
C. Jamerson
OGC-BETH
ACRS (10)
R. Diggs
T. Barnhart (8)
E. L. Jordan
J. Partlow
D. Hagan
J. Zwolinski
T. Rotella
E. Butcher
Wanda Jones
F. Miraglia
G. Zech, RII
B. Hayes
GPA/PA
SEQ Rdg

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 55
License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has 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 December 23, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 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 hereby amended by page changes to the Appendix A 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 Appendix A, as revised through Amendment No. 55 are hereby incorporated into 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

Original signed by

John A. Zwolinski, Assistant Director
for Projects
Division of TVA Projects
Office of Special Projects

Attachment
Appendix A Technical
Specification Changes

Date of Issuance: May 12, 1987

* SEE PREVIOUS CONCURRENCES

		PWR#4/DPWR-A *BKSingh 03/16/87	TVAPS/OSP JZwolinski 04/29/87
PWR#4/DPWR-A *MDuncan/rad 03/03/87	PWR#4/DPWR-A *JHolonich 03/03/87	OGC-BETH *SETurk 03/09/87	PWR#4/DPWR-A *BJYoungblood 03/16/87

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. are hereby incorporated into 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

B. J. Youngblood, Director
PWR Project Directorate #4
Division of PWR Licensing-A

Attachment"
Appendix A Technical
  Specification Changes

Date of Issuance:

PWR#4/DPWR-A	PWR#4/DPWR-A	OGC-BETH	PWR#4/DPWR-A
MDuncan/rad	JHolonich	SETurk	BJYoungblood
03/3/87	03/3/87	03/9/87	03/16/87

concurrence is subject to prior close of comment period w/out adverse comments.
3/9/87
3/6/87

ATTACHMENT TO LICENSE AMENDMENT NO. 55

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 contain vertical lines indicating the areas of change. Overleaf page provided to maintain document completeness.*

REMOVE	INSERT
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32*
3/4 3-33	3/4 3-33

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
1. Manual		
a.	Safety Injection (ECCS)	Not Applicable
	Feedwater Isolation	Not Applicable
	Reactor Trip (SI)	Not Applicable
	Containment Isolation-Phase "A"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
	Auxiliary Feedwater Pumps	Not Applicable
	Essential Raw Cooling Water System	Not Applicable
	Emergency Gas Treatment System	Not Applicable
b.	Containment Spray	Not Applicable
	Containment Isolation-Phase "B"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
	Containment Air Return Fan	Not Applicable
c.	Containment Isolation-Phase "A"	Not Applicable
	Emergency Gas Treatment System	Not Applicable
	Containment Ventilation Isolation	Not Applicable
d.	Steam Line Isolation	Not Applicable
2. Containment Pressure-High		
a.	Safety Injection (ECCS)	$\leq$ 32.0(1)
b.	Reactor Trip (from SI)	$\leq$ 3.0
c.	Feedwater Isolation	$\leq$ 8.0(2)
d.	Containment Isolation-Phase "A"(3)	$\leq$ 18.0(8)/28.0(9)
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	$\leq$ 60
g.	Essential Raw Cooling Water System	$\leq$ 65.0(8)/75.0(9)
h.	Emergency Gas Treatment System	$\leq$ 38.0(9)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 32.0(1)/28.0(7)
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$ 18.0(8)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Essential Raw Cooling Water System	$\leq$ 65.0(8)/75.0(9)
h. Emergency Gas Treatment System	$\leq$ 28.0(8)
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq$ 28.0(7)/28.0(1)
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$ 18.0(8)/28.0(9)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Essential Raw Cooling Water System	$\leq$ 65.0(8)/75.0(9)
h. Emergency Gas Treatment System	$\leq$ 38.0(9)
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	$\leq$ 30.0(7)/30.0(1)
b. Reactor Trip (from SI)	$\leq$ 5.0
c. Feedwater Isolation	$\leq$ 10.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$ 20.0(8)/30.0(9)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Essential Raw Cooling Water System	$\leq$ 67.0(8)/77.0(9)
h. Steam Line Isolation	$\leq$ 10.0
i. Emergency Gas Treatment System	$\leq$ 40.0(9)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 28.0(7)/28.0(1)
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$ 18.0(8)/28.0(9)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Essential Raw Cooling Water System	$\leq$ 65.0(8)/75.0(9)
h. Steam Line Isolation	$\leq$ 8.0
i. Emergency Gas Treatment System	$\leq$ 38.0(9)
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 58.00(9)
b. Containment Isolation-Phase "B"	$\leq$ 65(8)/75(9)
c. Steam Line Isolation	$\leq$ 7.0
d. Containment Air Return Fan	$\geq$ 540.0 and $\leq$ 660
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0(2)
9. Main Steam Generator Water Level - Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps(4)	$\leq$ 60.0
b. Turbine-driven Auxiliary Feedwater Pumps(5)	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Station Blackout	
a. Auxiliary Feedwater Pumps	≤ 60
11. Trip of Main Feedwater Pumps	
a. Auxiliary Feedwater Pumps	≤ 60
*12. Loss of Power	
a. 6.9 kv Shutdown Board - Degraded Voltage or Loss of Voltage	$\leq 10^{(10)}$
13. RWST Level-Low Coincident with Containment Sump Level-High and Safety Injection	
a. Automatic Switchover to Containment Sump	≤ 250
14. Containment Purge Air Exhaust Radioactivity - High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$
15. Containment Gas Monitor Radioactivity High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$
16. Containment Particulate Activity High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$

*NOTE: This technical specification to be implemented at the startup following the second refueling outage or following completion of the modification, whichever is earlier.

TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2) Using air operated valve

(3) The following valves are exceptions to the response times shown in the table and will have the values listed in seconds for the initiating signals and function indicated:

Valves: FCV-26-240, -243
Response times:
- 2.d. 21[8]/ 31[9]
- 3.d. 22[8]
- 4.d. 21[8]/ 31[9]
- 5.d. 24[8]/ 34[9]
- 6.d. 21[8]/31[9]

Valves: FCV-61-96, -97, -110, -122, -191, -192, -193, -194
Response times:
- 2.d. 31[8]
- 3.d. 32[8]
- 4.d. 31[8]
- 5.d. 34[8]
- 6.d. 31[8]

Valve: FCV-70-143
Response times:
- 2.d. 61[8]/71[9]
- 3.d. 62[8]
- 4.d. 61[8]/71[9]
- 5.d. 64[8]/74[9]
- 6.d. 61[8]/71[9]

(4) On 2/3 any Steam Generator

(5) On 2/3 in 2/4 Steam Generator

(6) Radiation detectors for Containment Ventilation Isolation may be excluded from Response Time Testing.

(7) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening and closing of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(8) Diesel generator starting and sequence loading delays not included. Response time limit includes operating time of valves.

(9) Diesel Generator starting and sequence loading delays included. Response time limit includes operating time of valves.

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. 47
License No. DPR-79

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

   A. The application for amendment to the Sequoyah Nuclear Plant, Unit 2 (the facility) Facility Operating License No. DPR-79 filed by the Tennessee Valley Authority (licensee), dated December 23, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 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 hereby amended by page changes to the Appendix A Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 47 are hereby incorporated into 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

Original signed by
John A. Zwolinski, Assistant Director
for Projects
Division of TVA Projects
Office of Special Projects

Attachment
Appendix A Technical
Specification Changes

Date of Issuance: May 12, 1987

* SEE PREVIOUS CONCURRENCES

		PWR#4/DPWR-A *BKSingh 03/16/87	TVAPS/OSP JZwolinski 4/29/87 04/29/87
PWR#4/DPWR-A *MDuncan/rad 03/03/87	PWR#4/DPWR-A *JHolonich 03/03/87	OGC-BETH *SETurk 03/09/87	PWR#4/DPWR-A *BJYoungblood 03/16/87

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.   are hereby incorporated into 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

B. J. Youngblood, Director
PWR Project Directorate #4
Division of PWR Licensing-A

Attachment
Appendix A Technical
Specification Changes

Date of Issuance:

PWR#4/DPWR-A	PWR#4/DPWR-A	OGC-Beth	PWR#4/DPWR-A
MDuncan/rad	JHolonich	SETurk	BJYoungblood
03/3/87	03/3/87	03/9/87	03/16/87

concurrence subject to prior close of comment period w/out adverse comment

WAsingh 3/10/87

3/9/87

ATTACHMENT TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. Overleaf page provided to maintain document completeness.*

REMOVE	INSERT
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32*
3/4 3-33	3/4 3-33

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
1. Manual		
a.	Safety Injection (ECCS)	Not Applicable
	Feedwater Isolation	Not Applicable
	Reactor Trip (SI)	Not Applicable
	Containment Isolation-Phase "A"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
	Auxiliary Feedwater Pumps	Not Applicable
	Essential Raw Cooling Water System	Not Applicable
	Emergency Gas Treatment System	Not Applicable
b.	Containment Spray	Not Applicable
	Containment Isolation-Phase "B"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
	Containment Air Return Fan	Not Applicable
c.	Containment Isolation-Phase "A"	Not Applicable
	Emergency Gas Treatment System	Not Applicable
	Containment Ventilation Isolation	Not Applicable
d.	Steam Line Isolation	Not Applicable
2. Containment Pressure-High		
a.	Safety Injection (ECCS)	$\leq$ 32.0(1)
b.	Reactor Trip (from SI)	$\leq$3.0
c.	Feedwater Isolation	$\leq$8.0(2)
d.	Containment Isolation-Phase "A"(3)	$\leq$18.0(8)/28.0(9)
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	$\leq$60
g.	Essential Raw Cooling Water System	$\leq$65.0(8)/75.0(9)
h.	Emergency Gas Treatment System	$\leq$38.0(9)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$32.0(1)/28.0(7)
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$18.0(8)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$60
g. Essential Raw Cooling Water System	$\leq$65.0(8)/75.0(9)
h. Emergency Gas Treatment System	$\leq$28.0(8)
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq$28.0(7)/28.0(1)
b. Reactor Trip (from SI)	$\leq$3.0
c. Feedwater Isolation	$\leq$8.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$18.0(8)/28.0(9)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$60
g. Essential Raw Cooling Water System	$\leq$65.0(8)/75.0(9)
h. Emergency Gas Treatment System	$\leq$38.0(9)
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	$\leq$30.0(7)/30.0(1)
b. Reactor Trip (from SI)	$\leq$5.0
c. Feedwater Isolation	$\leq$10.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$20.0(8)/30.0(9)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$60
g. Essential Raw Cooling Water System	$\leq$67.0(8)/77.0(9)
h. Steam Line Isolation	$\leq$10.0
i. Emergency Gas Treatment System	$\leq$40.0(9)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 28.0(7)/28.0(1)
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$ 18.0(8)/28.0(9)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$60
g. Essential Raw Cooling Water System	$\leq$ 65.0(8)/75.0(9)
h. Steam Line Isolation	$\leq$ 8.0
i. Emergency Gas Treatment System	$\leq$38.0(9)
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 58.00(9)
b. Containment Isolation-Phase "B"	$\leq$ 65(8)/75(9)
c. Steam Line Isolation	$\leq$ 7.0
d. Containment Air Return Fan	$\geq$ 540.0 and $\leq$660
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0(2)
9. Main Steam Generator Water Level - Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps(4)	$\leq$ 60.0
b. Turbine-driven Auxiliary Feedwater Pumps(5)	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Station Blackout	
a. Auxiliary Feedwater Pumps	$\leq$ 60
11. Trip of Main Feedwater Pumps	
a. Auxiliary Feedwater Pumps	$\leq$ 60
*12. Loss of Power	
a. 6.9 kv Shutdown Board - Degraded Voltage or Loss of Voltage	$\leq 10^{(10)}$
13. RWST Level-Low Coincident with Containment Sump Level-High and Safety Injection	
a. Automatic Switchover to Containment Sump	$\leq$ 250
14. Containment Purge Air Exhaust Radioactivity - High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$
15. Containment Gas Monitor Radioactivity High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$
16. Containment Particulate Activity High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$

*NOTE: This technical specification is to be implemented during the startup following the first refueling outage.

TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2) Using air operated valve

(3) The following valves are exceptions to the response times shown in the table and will have the values listed in seconds for the initiating signals and function indicated:

Valves: FCV-26-240, -243
Response times: 2.d. $21^{(8)}/31^{(9)}$
3.d. $22^{(8)}$
4.d. $21^{(8)}/31^{(9)}$
5.d. $24^{(8)}/34^{(9)}$
6.d. $21^{(8)}/31^{(9)}$

Valves: FCV61-96, -97, -110, -122, -191, -192, -193, -194
Response times
2.d. $31^{(8)}$
3.d. $32^{(8)}$
4.d. $31^{(8)}$
5.d. $34^{(8)}$
6.d. $31^{(8)}$

Valve: FCV-70-143
Response times: 2.d. $61^{(8)}/71^{(9)}$
3.d. $62^{(8)}$
4.d. $61^{(8)}/71^{(9)}$
5.d. $64^{(8)}/74^{(9)}$
6.d. $61^{(8)}/71^{(9)}$

(4) On 2/3 any Steam Generator

(5) On 2/3 in 2/4 Steam Generator

(6) Radiation detectors for Containment Ventilation Isolation may be excluded from Response Time Testing.

(7) Diesel generator starting and sequence loading delays <u>not</u> included. Offsite power available. Response time limit includes opening and closing of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(8) Diesel generator starting and sequence loading delays not included. Response time limit includes operating time of valves.

(9) Diesel generator starting and sequence loading delays included. Response time limit includes operating time of valves.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE DPR-77

AND AMENDMENT NO. 47 TO FACILITY OPERATING LICENSE DPR-79

TENNESSEE VALLEY AUTHORITY

INTRODUCTION

By letter dated December 23, 1986, the licensee requested technical specification changes that would permit installation of slower acting valve operators on valves between the suction of the centrifugal charging pumps and the volume control tank. The valves are designed to close following a safety injection signal to permit the centrifugal charging (CC) pumps to take suction solely from the refueling water storage tank (RWST). Closure of the valves is required since full flow from the centrifugal charging pumps would cause the volume control tank (VCT) to be drained within a few minutes. In addition closure of the valves ensures that the 2000 ppm boric acid solution in the RWST will be pumped into the core to terminate the reactivity excursion which would be produced by a large steam line break. The boric acid concentration in the VCT is normally the same as the reactor system.

EVALUATION

The valves between the VCT and the CC pumps suction are currently equipped with fast closing operators which require a device to limit the impact of the valve internals on the valve seats. Motor breaks were therefore installed for the valves at both Units 1 and 2. The motor breaks at Unit 1 are currently inoperable. TVA has determined that the installation of slower closing valve operators will be acceptable and proposes to modify the operators at both units to decrease the closing speed and to remove the motor breaks at both units. The use of slower valve operators would increase the closing time by 15 seconds over that which was assumed in the FSAR safety evaluations and requires that the safety injection delay times in the Technical Specifications be increased by 5 seconds.

TVA determined that the consequences of design basis transients and accidents would be unaffected by the slower closing of the valves. The licensee presented that Westinghouse performed an evaluation of delayed ECCS actuation consistent with the increased response times in the Technical Specifications. It was determined from that evaluation that delayed ECCS actuation has the potential of affecting the mitigation of steam line break and LOCA events. Following a large break LOCA the accumulators would inject abundant ECCS water into the reactor during the blowdown period when the valves would be closing. Before the reflooding period when flow from the CC pumps aids in reflooding the core, the CC pump suction valves would have closed so that the time to recover the core would be unaffected. In the case of a small break LOCA the peak cladding temperature is not reached for over 1000 seconds after the break occurs. An additional delay in ECCS response as requested for the Technical Specifications would have an insignificant effect on the total pumped flow and hence on the amount of core heatup. In the case of a break in a main steam line, previously approved analyses for removal of the boron injection tank (BIT) concluded that recriticality and return to power during blowdown of one steam generator would not cause fuel failure. Injection of boric acid from the RWST would be required eventually to terminate the event; however sensitivity studies by Westinghouse have determined that the requested delay in the ECCS response would not have a significant effect on the minimum DNBR which would occur at approximately 100 seconds into the event. Based on the above considerations the staff concludes that the Technical Specifications for Sequoyah Units 1 and 2 may be modified as requested by the licensee.

ENVIRONMENTAL CONSIDERATION

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

CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register on February 26, 1987 (52 FR 5870) and consulted with the state of Tennessee. No public comments were received, and the state of Tennessee did not have any comments.

We have 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Joe Holonich, PWR#4, DPWR-A
Walt Jensen, PARS, DPWR-A

Dated: May 12, 1987

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 55
License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has 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 December 23, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 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 hereby amended by page changes to the Appendix A 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:

~~8705260044~~ 18pp

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 55 are hereby incorporated into 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

John A. Zwolinski, Assistant Director
for Projects
Division of TVA Projects
Office of Special Projects

Attachment
Appendix A Technical
Specification Changes

Date of Issuance: May 12, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 55

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 contain vertical lines indicating the areas of change. Overleaf page provided to maintain document completeness.*

REMOVE	INSERT
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32*
3/4 3-33	3/4 3-33

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Ventilation Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Essential Raw Cooling Water System	Not Applicable
Emergency Gas Treatment System	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Ventilation Isolation	Not Applicable
Containment Air Return Fan	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Emergency Gas Treatment System	Not Applicable
Containment Ventilation Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 32.0^{(1)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	$\leq 8.0^{(2)}$
d. Containment Isolation-Phase "A"(3)	$\leq 18.0^{(8)}/28.0^{(9)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Essential Raw Cooling Water System	$\leq 65.0^{(8)}/75.0^{(9)}$
h. Emergency Gas Treatment System	$\leq 38.0^{(9)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 32.0(1)/28.0(7)
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$ 18.0(8)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Essential Raw Cooling Water System	$\leq$ 65.0(8)/75.0(9)
h. Emergency Gas Treatment System	$\leq$ 28.0(8)
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq$ 28.0(7)/28.0(1)
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$ 18.0(8)/28.0(9)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Essential Raw Cooling Water System	$\leq$ 65.0(8)/75.0(9)
h. Emergency Gas Treatment System	$\leq$ 38.0(9)
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	$\leq$ 30.0(7)/30.0(1)
b. Reactor Trip (from SI)	$\leq$ 5.0
c. Feedwater Isolation	$\leq$ 10.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$ 20.0(8)/30.0(9)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Essential Raw Cooling Water System	$\leq$ 67.0(8)/77.0(9)
h. Steam Line Isolation	$\leq$ 10.0
i. Emergency Gas Treatment System	$\leq$ 40.0(9)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 28.0(7)/28.0(1)
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$ 18.0(8)/28.0(9)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Essential Raw Cooling Water System	$\leq$ 65.0(8)/75.0(9)
h. Steam Line Isolation	$\leq$ 8.0
i. Emergency Gas Treatment System	$\leq$ 38.0(9)
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 58.00(9)
b. Containment Isolation-Phase "B"	$\leq$ 65(8)/75(9)
c. Steam Line Isolation	$\leq$ 7.0
d. Containment Air Return Fan	$\geq$ 540.0 and $\leq$ 660
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0(2)
9. Main Steam Generator Water Level - Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps(4)	$\leq$ 60.0
b. Turbine-driven Auxiliary Feedwater Pumps(5)	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Station Blackout	
a. Auxiliary Feedwater Pumps	$\leq$ 60
11. Trip of Main Feedwater Pumps	
a. Auxiliary Feedwater Pumps	$\leq$ 60
*12. Loss of Power	
a. 6.9 kv Shutdown Board - Degraded Voltage or Loss of Voltage	$\leq 10^{(10)}$
13. RWST Level-Low Coincident with Containment Sump Level-High and Safety Injection	
a. Automatic Switchover to Containment Sump	$\leq$ 250
14. Containment Purge Air Exhaust Radioactivity - High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$
15. Containment Gas Monitor Radioactivity High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$
16. Containment Particulate Activity High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$

*NOTE: This technical specification to be implemented at the startup following the second refueling outage or following completion of the modification, whichever is earlier.

TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2) Using air operated valve

(3) The following valves are exceptions to the response times shown in the table and will have the values listed in seconds for the initiating signals and function indicated:

Valves: FCV-26-240, -243
Response times:
2.d. 21[8]/ 31[9]
3.d. 22[8]
4.d. 21[8]/ 31[9]
5.d. 24[8]/ 34[9]
6.d. 21[8]/31[9]

Valves: FCV-61-96, -97, -110, -122, -191, -192, -193, -194
Response times:
2.d. 31[8]
3.d. 32[8]
4.d. 31[8]
5.d. 34[8]
6.d. 31[8]

Valve: FCV-70-143
Response times:
2.d. 61[8]/71[9]
3.d. 62[8]
4.d. 61[8]/71[9]
5.d. 64[8]/74[9]
6.d. 61[8]/71[9]

(4) On 2/3 any Steam Generator

(5) On 2/3 in 2/4 Steam Generator

(6) Radiation detectors for Containment Ventilation Isolation may be excluded from Response Time Testing.

(7) Diesel generator starting and sequence loading delays <u>not</u> included. Offsite power available. Response time limit includes opening and closing of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(8) Diesel generator starting and sequence loading delays not included. Response time limit includes operating time of valves.

(9) Diesel Generator starting and sequence loading delays included. Response time limit includes operating time of valves.

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. 47
License No. DPR-79

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

   A. The application for amendment to the Sequoyah Nuclear Plant, Unit 2 (the facility) Facility Operating License No. DPR-79 filed by the Tennessee Valley Authority (licensee), dated December 23, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 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 hereby amended by page changes to the Appendix A Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 47 are hereby incorporated into 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

John A. Zwolinski, Assistant Director
for Projects
Division of TVA Projects
Office of Special Projects

Attachment
Appendix A Technical
Specification Changes

Date of Issuance: May 12, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. Overleaf page provided to maintain document completeness.*

REMOVE	INSERT
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32*
3/4 3-33	3/4 3-33

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
1. Manual		
a.	Safety Injection (ECCS)	Not Applicable
	Feedwater Isolation	Not Applicable
	Reactor Trip (SI)	Not Applicable
	Containment Isolation-Phase "A"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
	Auxiliary Feedwater Pumps	Not Applicable
	Essential Raw Cooling Water System	Not Applicable
	Emergency Gas Treatment System	Not Applicable
b.	Containment Spray	Not Applicable
	Containment Isolation-Phase "B"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
	Containment Air Return Fan	Not Applicable
c.	Containment Isolation-Phase "A"	Not Applicable
	Emergency Gas Treatment System	Not Applicable
	Containment Ventilation Isolation	Not Applicable
d.	Steam Line Isolation	Not Applicable
2. Containment Pressure-High		
a.	Safety Injection (ECCS)	$\leq$ 32.0(1)
b.	Reactor Trip (from SI)	$\leq$3.0
c.	Feedwater Isolation	$\leq$8.0(2)
d.	Containment Isolation-Phase "A"(3)	$\leq$18.0(8)/28.0(9)
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	$\leq$60
g.	Essential Raw Cooling Water System	$\leq$65.0(8)/75.0(9)
h.	Emergency Gas Treatment System	$\leq$38.0(9)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 32.0^{(1)}/28.0^{(7)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	$\leq 8.0^{(2)}$
d. Containment Isolation-Phase "A"(3)	$\leq 18.0^{(8)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Essential Raw Cooling Water System	$\leq 65.0^{(8)}/75.0^{(9)}$
h. Emergency Gas Treatment System	$\leq 28.0^{(8)}$
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq 28.0^{(7)}/28.0^{(1)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	$\leq 8.0^{(2)}$
d. Containment Isolation-Phase "A"(3)	$\leq 18.0^{(8)}/28.0^{(9)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Essential Raw Cooling Water System	$\leq 65.0^{(8)}/75.0^{(9)}$
h. Emergency Gas Treatment System	$\leq 38.0^{(9)}$
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	$\leq 30.0^{(7)}/30.0^{(1)}$
b. Reactor Trip (from SI)	≤ 5.0
c. Feedwater Isolation	$\leq 10.0^{(2)}$
d. Containment Isolation-Phase "A"(3)	$\leq 20.0^{(8)}/30.0^{(9)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Essential Raw Cooling Water System	$\leq 67.0^{(8)}/77.0^{(9)}$
h. Steam Line Isolation	≤ 10.0
i. Emergency Gas Treatment System	$\leq 40.0^{(9)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 28.0(7)/28.0(1)
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$ 18.0(8)/28.0(9)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$60
g. Essential Raw Cooling Water System	$\leq$ 65.0(8)/75.0(9)
h. Steam Line Isolation	$\leq$ 8.0
i. Emergency Gas Treatment System	$\leq$38.0(9)
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 58.00(9)
b. Containment Isolation-Phase "B"	$\leq$ 65(8)/75(9)
c. Steam Line Isolation	$\leq$ 7.0
d. Containment Air Return Fan	$\geq$ 540.0 and $\leq$660
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0(2)
9. Main Steam Generator Water Level - Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps(4)	$\leq$ 60.0
b. Turbine-driven Auxiliary Feedwater Pumps(5)	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Station Blackout	
a. Auxiliary Feedwater Pumps	≤ 60
11. Trip of Main Feedwater Pumps	
a. Auxiliary Feedwater Pumps	≤ 60
*12. Loss of Power	
a. 6.9 kv Shutdown Board - Degraded Voltage or Loss of Voltage	$\leq 10^{(10)}$
13. RWST Level-Low Coincident with Containment Sump Level-High and Safety Injection	
a. Automatic Switchover to Containment Sump	≤ 250
14. Containment Purge Air Exhaust Radioactivity - High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$
15. Containment Gas Monitor Radioactivity High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$
16. Containment Particulate Activity High	
a. Containment Ventilation Isolation	$\leq 10^{(6)}$

*NOTE: This technical specification is to be implemented during the startup following the first refueling outage.

TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2) Using air operated valve

(3) The following valves are exceptions to the response times shown in the table and will have the values listed in seconds for the initiating signals and function indicated:

Valves: FCV-26-240, -243
Response times:
2.d. 21[8]/31[9]
3.d. 22[8]
4.d. 21[8]/31[9]
5.d. 24[8]/34[9]
6.d. 21[8]/31[9]

Valves: FCV61-96, -97, -110, -122, -191, -192, -193, -194
Response times
2.d. 31[8]
3.d. 32[8]
4.d. 31[8]
5.d. 34[8]
6.d. 31[8]

Valve: FCV-70-143
Response times:
2.d. 61[8]/71[9]
3.d. 62[8]
4.d. 61[8]/71[9]
5.d. 64[8]/74[9]
6.d. 61[8]/71[9]

(4) On 2/3 any Steam Generator

(5) On 2/3 in 2/4 Steam Generator

(6) Radiation detectors for Containment Ventilation Isolation may be excluded from Response Time Testing.

(7) Diesel generator starting and sequence loading delays <u>not</u> included. Offsite power available. Response time limit includes opening and closing of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(8) Diesel generator starting and sequence loading delays not included. Response time limit includes operating time of valves.

(9) Diesel generator starting and sequence loading delays included. Response time limit includes operating time of valves.

UNITED STATES
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WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.55 TO FACILITY OPERATING LICENSE DPR-77

AND AMENDMENT NO.47 TO FACILITY OPERATING LICENSE DPR-79

TENNESSEE VALLEY AUTHORITY

INTRODUCTION

By letter dated December 23, 1986, the licensee requested technical specification changes that would permit installation of slower acting valve operators on valves between the suction of the centrifugal charging pumps and the volume control tank. The valves are designed to close following a safety injection signal to permit the centrifugal charging (CC) pumps to take suction solely from the refueling water storage tank (RWST). Closure of the valves is required since full flow from the centrifugal charging pumps would cause the volume control tank (VCT) to be drained within a few minutes. In addition closure of the valves ensures that the 2000 ppm boric acid solution in the RWST will be pumped into the core to terminate the reactivity excursion which would be produced by a large steam line break. The boric acid concentration in the VCT is normally the same as the reactor system.

EVALUATION

The valves between the VCT and the CC pumps suction are currently equipped with fast closing operators which require a device to limit the impact of the valve internals on the valve seats. Motor breaks were therefore installed for the valves at both Units 1 and 2. The motor breaks at Unit 1 are currently inoperable. TVA has determined that the installation of slower closing valve operators will be acceptable and proposes to modify the operators at both units to decrease the closing speed and to remove the motor breaks at both units. The use of slower valve operators would increase the closing time by 15 seconds over that which was assumed in the FSAR safety evaluations and requires that the safety injection delay times in the Technical Specifications be increased by 5 seconds.

TVA determined that the consequences of design basis transients and accidents would be unaffected by the slower closing of the valves. The licensee presented that Westinghouse performed an evaluation of delayed ECCS actuation consistent with the increased response times in the Technical Specifications. It was determined from that evaluation that delayed ECCS actuation has the potential of affecting the mitigation of steam line break and LOCA events. Following a large break LOCA the accumulators would inject abundant ECCS water into the reactor during the blowdown period when the valves would be closing. Before the reflooding period when flow from the CC pumps aids in reflooding the core, the CC pump suction valves would have closed so that the time to recover the core would be unaffected. In the case of a small break LOCA the peak cladding temperature is not reached for over 1000 seconds after the break occurs. An additional delay in ECCS response as requested for the Technical Specifications would have an insignificant effect on the total pumped flow and hence on the amount of core heatup. In the case of a break in a main steam line, previously approved analyses for removal of the boron injection tank (BIT) concluded that recriticality and return to power during blowdown of one steam generator would not cause fuel failure. Injection of boric acid from the RWST would be required eventually to terminate the event; however sensitivity studies by Westinghouse have determined that the requested delay in the ECCS response would not have a significant effect on the minimum DNBR which would occur at approximately 100 seconds into the event. Based on the above considerations the staff concludes that the Technical Specifications for Sequoyah Units 1 and 2 may be modified as requested by the licensee.

ENVIRONMENTAL CONSIDERATION

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

CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register on February 26, 1987 (52 FR 5870) and consulted with the state of Tennessee. No public comments were received, and the state of Tennessee did not have any comments.

We have 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Joe Holonich, PWR#4, DPWR-A
Walt Jensen, PARS, DPWR-A

Dated: May 12, 1987

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

May 12, 1987

Docket Nos. 50-237/238

MEMORANDUM FOR: Sholly Coordinator

FROM: John A. Zwolinski, Assistant Director for Projects, OSP

SUBJECT: REQUEST FOR PUBLICATION IN BI-WEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Tennessee Valley Authority, Docket Nos. 50-237 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: December 23, 1986

Brief description of amendments: The amendments change the Technical Specifications to permit installation of slower acting valve operators on valves between the suction of the centrifugal charging pumps and the volume control tank.

Date of issuance: May 12, 1987

Effective date: May 12, 1987

Amendment Nos.: 55 and 47

Facility Operating License Nos. DPR-77 and DPR-79. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 26,1987 (52 FR 5870)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 12, 1987 .

No significant hazards consideration comments received: No

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Original signed by
John A. Zwolinski, Assistant Director
for Projects, OSP

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