May 4, 1982

Docket Nos: 50-327
and 50-328

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. 13 to Facility Operating License No. DPR-77 and Amendment No. 4 to Facility Operating License No. DPR-79 - Sequoyah Nuclear Plant, Units 1 and 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 13 to Facility Operating License No. DPR-77 and Amendment No. 4 to Facility Operating License No. DPR-79.

These amendments change various sections of the Technical Specifications related to fire hose test pressure requirements, vital battery surveillance, ice bed temperature determination, isolation valve closing times, downscale failure alarms, visual inspection schedule on snubbers, maximum enrichment for reload fuel, quality assurance monitoring of plant effluents, land use census, and system flushing.

A copy of the related safety evaluation supporting Amendment No. 13 to Facility Operating License DPR-77 and Amendment No. 4 to Facility Operating License DPR-79 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. CP

Sincerely,

S/

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

Enclosures:
1. Amendment No. 13 to DPR-77
2. Amendment No. 4 to DPR-79
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
See next page

OFFICE		DL:LB #4:LA	DL:LB #4	DL:LB #4			
SURNAME		MDuncan/hmc	CStahle	EAdensam			
DATE		4/14/82	4/15/82	4/30/82			

SEQUOYAH

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

cc: Herbert S. Sanger, Jr., Esq.
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E 11B 33
Knoxville, Tennessee 37902

Mr. H. N. Culver
Tennessee Valley Authority
400 Commerce Avenue, 249A HBB
Knoxville, Tennessee 37902

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

Mr. Mark Burzynski
Tennessee Valley Authority
400 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Mr. J. F. Cox
Tennessee Valley Authority
400 Commerce Avenue, W10C131C
Knoxville, Tennessee 37902

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

Director, Office of Urban
& Federal Affairs
108 Parkway Towers
404 James Robertson Way
Nashville, Tennessee 37219

Attorney General
Supreme Court Building
Nashville, Tennessee 37219

U.S. Environmental Protection
Agency
ATTN: EIS Coordinator
345 Courtland Street
Atlanta, Georgia 30308

Honorable Don Moore, Jr.
County Judge
Hamilton County Courthouse
Chattanooga, Tennessee 37402

Regional Administrator
Nuclear Regulatory Commission,
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

OFFICE ▶							
SURNAME ▶							
DATE ▶							

<u>TENNESSEE VALLEY AUTHORITY</u>

<u>DOCKET NO. 50-327</u>

<u>SEQUOYAH NUCLEAR PLANT, UNIT 1</u>

<u>AMENDMENT TO FACILITY OPERATING LICENSE</u>

Amendment No. 13
License No. DPR-77

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

   A. The applications 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 10, 1981 (two letters), March 1, March 9, and April 7, 1982, comply 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 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 attachments to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 13, 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

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

Attachment:
Appendix A Technical
Specification Changes

Date of Issuance: May 4, 1982

OFFICE	LA:DL:LB #4	DL:LB #4	OELD	DL:LB #4	AD:L
SURNAME	MDuncan/hmc	CStahle	[illegible]	EAdensam	RTedesco
DATE	4/14/82	4/15/82	4/16/82	4/30/82	05/03/82

ATTACHMENT TO LICENSE AMENDMENT NO. 13

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 area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Overleaf Page	Amended Page
	1-8
	3/4 3-37
	3/4 3-73
	3/4 3-80
	3/4 6-19
	3/4 6-20
	3/4 6-28
	3/4 7-21
	3/4 7-31
	3/4 7-37
	3/4 7-38
	3/4 7-39
	3/4 7-40
3/4 12-9	3/4 12-10
	5-5
	6-15
	6-15a
5-3	5-4

In order to clarify changes made in Amendment No. 12, please delete the following pages:

1-9
3/4 6-25b

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Completed prior to each release.
N.A.	Not applicable.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
c. Main Steam Generator Water Level-Low-Low	S	R	M	1, 2, 3
d. S.I.	See 1 above (all SI surveillance requirements)			
e. Station Blackout	N.A.	R	N.A.	1, 2, 3
f. Trip of Main Feedwater Pumps	N.A.	N.A.	R	1, 2
g. Auxiliary Feedwater Suction Pressure - Low	N.A.	R	M	1, 2, 3
7. LOSS OF POWER				
a. 6.9 kv Shutdown Board Undervoltage				
1. Loss of Voltage	S	R	M	1, 2, 3, 4
2. Load Shedding	S	R	N.A.	1, 2, 3, 4
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS				
a. Pressurizer Pressure, P-11	N.A.	R (4)	N.A.	1, 2, 3
b. T_{avg}, P-12	N.A.	R (4)	N.A.	1, 2, 3
c. Steam Generator Level, P-14	N.A.	R (4)	N.A.	1, 2
9. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP				
a. RSWT Level - Low	S	R	M	1, 2, 3, 4
COINCIDENT WITH				
Containment Sump Level - High	S	R	M	1, 2, 3, 4
AND				
Safety Injection	(See 1 above for all Safety Injection Surveillance Requirements)			

TABLE 4.3.8 (Continued)

TABLE NOTATION

* During liquid additions to the tank.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

TABLE 4.3-9 (Continued)

TABLE NOTATION

* At all times.

** During waste gas disposal system operation.

*** During shield building exhaust system operation.

**** During waste gas releases.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure.

(2) The CHANNEL FUNCTION TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

TABLE 3.6-2

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
A.	PHASE "A" ISOLATION		
1.	FCV-1-7	SG Blow Dn	10
2.	FCV-1-14	SG Blow Dn	10
3.	FCV-1-25	SG Blow Dn	10
4.	FCV-1-32	SG Blow Dn	10
5.	FCV-1-181	SG Blow Dn	15
6.	FCV-1-182	SG Blow Dn	15
7.	FCV-1-183	SG Blow Dn	15
8.	FCV-1-184	SG Blow Dn	15
9.	FCV-31C-222	CW-Inst Room Clrs	10
10.	FCV-31C-223	CW-Inst Room Clrs	10
11.	FCV-31C-224	CW-Inst Room Clrs	10
12.	FCV-31C-225	CW-Inst Room Clrs	10
13.	FCV-31C-229	CW-Inst Room Clrs	10
14.	FCV-31C-230	CW-Inst Room Clrs	10
15.	FCV-31C-231	CW-Inst Room Clrs	10
16.	FCV-31C-232	CW-Inst Room Clrs	10
17.	FCV-43-22	Sample RC Outlet Hdrs	10
18.	FCV-43-23	Sample RC Outlet Hdrs	10
19.	FCV-43-55	SG Blow Dn Sample Line	10
20.	FCV-43-58	SG Blow Dn Sample Line	10
21.	FCV-43-61	SG Blow Dn Sample Line	10
22.	FCV-43-64	SG Blow Dn Sample Line	10
23.	FCV-61-96	Gylcol Inlet to Floor Cooler	30
24.	FCV-61-97	Gylcol Inlet to Floor Cooler	30
25.	FCV-61-110	Gylcol Outlet to Floor Cooler	30
26.	FCV-61-122	Gylcol Outlet to Floor Cooler	30
27.	FCV-61-191	Ice Condenser - Gylcol In	30
28.	FCV-61-192	Ice Condenser - Gylcol In	30
29.	FCV-61-193	Ice Condenser - Gylcol Out	30
30.	FCV-61-194	Ice Condenser - Gylcol Out	30
31.	FCV-62-61	RCP Seals	10

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
A.	PHASE "A" ISOLATION (Cont.)		
32.	FCV-62-63	RCP Seals	10
33.	FCV-62-72	Letdown Line	10
34.	FCV-62-73	Letdown Line	10
35.	FCV-62-74	Letdown Line	10
36.	FCV-62-77	Letdown Line	10
37.	FCV-63-23	Accum to Hold Up Tank	10
38.	FCV-63-64	WDS N_2 to Accum	10
39.	FCV-63-71	Accum to Hold Up Tank	10
40.	FCV-63-84	Accum to Hold Up Tank	10
41.	FCV-68-305	WDS N_2 to PRT	10
42.	FCV-68-307	PRT to Gas Analyzer	10
43.	FCV-68-308	PRT to Gas Analyzer	10
44.	FCV-70-85	CCS from Excess Lt Dn Hx	10
45.	FCV-70-143	CCS to Excess Lt Dn Hx	60
46.	FCV-77-9	RCDT Pump Disch	10
47.	FCV-77-10	RCDT Pump Disch	10
48.	FCV-77-16	RCDT to Gas Analyzer	10
49.	FCV-77-17	RCDT to Gas Analyzer	10
50.	FCV-77-18	RCDT and PRT to V H	10
51.	FCV-77-19	RCDT and PRT to V H	10
52.	FCV-77-20	N_2 to RCDT	10
53.	FCV-77-127	Floor Sump Pump Disch	10
54.	FCV-77-128	Floor Sump Pump Disch	10
55.	FCV-81-12	Primary Water Makeup	10
56.	FCV-87-7	UHI Test Line	10
57.	FCV-87-8	UHI Test Line	10
58.	FCV-87-9	UHI Test Line	10
59.	FCV-87-10	UHI Test Line	10
60.	FCV-87-11	UHI Test Line	10
61.	FCV-26-240	Fire Protection Isol.	20
62.	FCV-26-243	Fire Protection Isol.	20

CONTAINMENT SYSTEMS

ICE BED TEMPERATURE MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.2 The ice bed temperature monitoring system shall be OPERABLE with at least 2 OPERABLE RTD channels in the ice bed at each of 3 basic elevations (10'6", 30'9" and 55' above the floor of the ice condenser) for each one third of the ice condenser.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the ice bed temperature indication not available in the main control room, determine the ice bed temperature at the local ice condenser temperature monitoring panel every 12 hours

b. With the ice bed temperature monitoring system inoperable, and unable to determine ice bed temperature at the local panel, POWER OPERATION may continue for up to 30 days provided:

   1. The ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed;

   2. The last recorded mean ice bed temperature was less than or equal to 20°F and steady; and

   3. The ice condenser cooling system is OPERABLE with at least:

      a) 21 OPERABLE air handling units,

      b) 2 OPERABLE glycol circulating pumps, and

      c) 3 OPERABLE refrigerant units;

   otherwise, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the ice bed temperature monitoring system inoperable and unable to determine the ice bed temperatures by alternate means and with the ice condenser cooling system not satisfying the minimum components OPERABILITY requirements of a.3 above, POWER OPERATION may continue for up to 6 days provided the ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed and the last recorded mean ice bed temperature was less than or equal to 15°F and steady; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 The ice bed temperature monitoring system shall be determined OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours.

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9. All safety-related snubbers shall be OPERABLE. The snubbers are shown in Tables 4.7.9.a and 4.7.9.b and are listed in Surveillance Instruction SNP SI-162. Any exemptions to the surveillance program are shown in Table 4.7.9.c and in SNP SI-162.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems or partial systems required OPERABLE in those MODES.)

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9. Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5. These snubbers are shown in Tables 4.7.9.a and 4.7.9.b, and are listed in Surveillance Instruction SNP SI-162. Table 4.7.9.b is a detailed tabulation of the hydraulic snubbers which are also shown in Table 4.7.9.a. Any exemption to any portion of the surveillance program for any snubber is shown in Table 4.7.9.c.

a. Inspection Groups

The snubbers may be categorized into two major groups based on whether the snubbers are accessible or inaccessible during reactor operation. These major groups may be further subdivided into subgroups based on design, environment, or other features which may be expected to affect the OPERABILITY of the snubbers within the subgroup. Each subgroup or group may be inspected independently in accordance with 4.7.9.b through 4.7.9.h.

b. Visual Inspection Schedule and Lot Size

The first inservice visual inspection of snubbers shall be completed by October 31, 1981, and shall include all snubbers on safety-related systems. If less than two (2) snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 18 months ± 25% from the date of the first inspection or during an outage of sufficient duration (at least 72 hours in Mode 5). Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

ACTION: (Continued)

c) In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.11.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by starting each electric motor driven pump and operating it for at least 15 minutes on recirculation flow.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

* c. At least once per 6 months by performance of a system flush.

d. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

e. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that each automatic valve in the flow path actuates to its correct position,

2. Verifying that each pump develops at least 1174 gpm at a system head of 312 feet,

3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

4. Verifying that the No. 1 fire pump starts to maintain the fire suppression water system pressure greater than or equal to 125 psig and that the No. 2 fire pump also starts automatically within 10 $\pm$ 2 seconds when the fire suppression water system is not maintained greater than or equal to 125 psig by the No. 1 pump.

f. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

*Note: These flushes should coincide with the chlorination of the raw service and fire suppression water system. These flushes should be run, one between April 1 and June 30, and the other between September 1 and November 15.

Within the prescribed spring and fall test period, deviation from the six-month performance frequency is authorized.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.4 The fire hose stations shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-5 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours. Restore the fire hose station to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the station to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the fire hose stations shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operations to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

   2. Removing the hose for inspection and re-racking, and

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

TABLE 3.7-5

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
a. Reactor Building - Annulus Area		
Platform	778.5	1-26-1196
Platform	778.5	1-26-1197
Platform	778.5	1-26-1198
Platform	778.5	1-26-1199
Platform	759.5	1-26-1200
Platform	759.5	1-26-1201
Platform	759.5	1-26-1202
Platform	759.5	1-26-1203
Platform	740.5	1-26-1204
Platform	740.5	1-26-1205
Platform	740.5	1-26-1206
Platform	740.5	1-26-1207
Platform	721.5	1-26-1208
Platform	721.5	1-26-1209
Platform	721.5	1-26-1210
Platform	721.5	1-26-1211
Platform	701.5	1-26-1212
Platform	701.5	1-26-1213
Platform	701.5	1-26-1214
Platform	701.5	1-26-1215
Platform	679.78	1-26-1216
Platform	679.78	1-26-1217
Platform	679.78	1-26-1218
Platform	679.78	1-26-1219
b. Reactor Building - RCP & Lower Containment Air Filters Area		
Reactor Building	679.78	1-26-1220
Reactor Building	679.78	1-26-1221
Reactor Building	679.78	1-26-1222
Reactor Building	679.78	1-26-1223
Reactor Building	679.78	1-26-1224
Reactor Building	679.78	1-26-1225
c. Control Building		
Control Building	732	0-26-1186
Control Building	732	0-26-1191
Control Building	706	0-26-1187
Control Building	706	0-26-1192

TABLE 3.7-5 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION		HOSE RACK#
Control Building	685		0-26-1188
Control Building	685		0-26-1193
Control Building	669		0-26-1189
Control Building	669		0-26-1194
d. Diesel Generator Building			
Corridor	722		0-26-1077
Corridor	740.5		0-26-1080
Air Exhaust Rm.			0-26-1082
e. Additional Equipment Building - Unit 1			
South Wall	740.5		1-26-687
South Wall	706		1-26-686
f. Auxiliary Building			
	759		1-26-669
	749		2-26-664
	749		1-26-664
	734		2-26-670
	734		0-26-684
	734		1-26-670
	734		0-26-682
	734	Siamese Outlet	1-26-671
	734		1-26-672
	734		1-26-665
	714		0-26-660
	714		1-26-666
	714		0-26-677
	706		0-26-658
	690		0-26-690
	690		0-26-661
	690	Siamese Outlet	1-26-674
	690		1-26-675
	690		1-26-667
	669		1-26-668
	669		0-26-662
	669		0-26-680
	653		0-26-663
	653		0-26-691

TABLE 3.7-5 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
g. CCW Intake Pumping Station		
	690	0-26-866
	690	0-26-867
	690	0-26-868
	690	0-26-869
	690	0-26-870
h. ERCW Pumping Station		
	688	0-26-927
	688	0-26-926
	688	0-26-930
	704	0-26-931
	704	0-26-925
	704	0-26-928
	720	0-26-929
	720	0-26-924
	720	0-26-932

TABLE NOTATION

b. The LLD for analysis of drinking water and surface water samples shall be performed by gamma spectroscopy at approximately 15 pCi/ℓ. If levels greater than 15 pCi/ℓ are identified in surface water samples downstream from the plant, or in the event of an unanticipated release of I-131, drinking water samples will be analyzed at a LLD of 1.0 pCi/ℓ for I-131.

c. Other peaks which are measurable and identifiable, together with the radionuclides in Table 4.12-1, shall be identified and reported.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of three miles.)

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the maximum values currently being calculated in Specification 4.11.2.3.1, in lieu of any other report required by Specification 6.9.1., prepare and submit to the Commission within 30 days after the updated calculation, pursuant to Specification 6.9.2, a Special Report which identifies the new location(s).

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than the highest calculated dose or dose commitment at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days after the updated calculation, pursuant to Specification 6.9.2, a Special Report which identifies the new location. The new location shall be added to the radiological environmental monitoring program within 30 days, if the owner consents. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per calendar year between the dates of April 1 and October 1 using the following techniques:

1. Within a 2-mile radius from the plant or within the 15-mrem/year isodose line, whichever is larger, enumeration by a door-to-door or equivalent counting technique.

2. Within a 5-mile radius from the plant, enumeration by using appropriate techniques such as door-to-door survey, mail survey, telephone survey, aerial survey, or information from local agricultural authorities or other reliable sources.

*Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

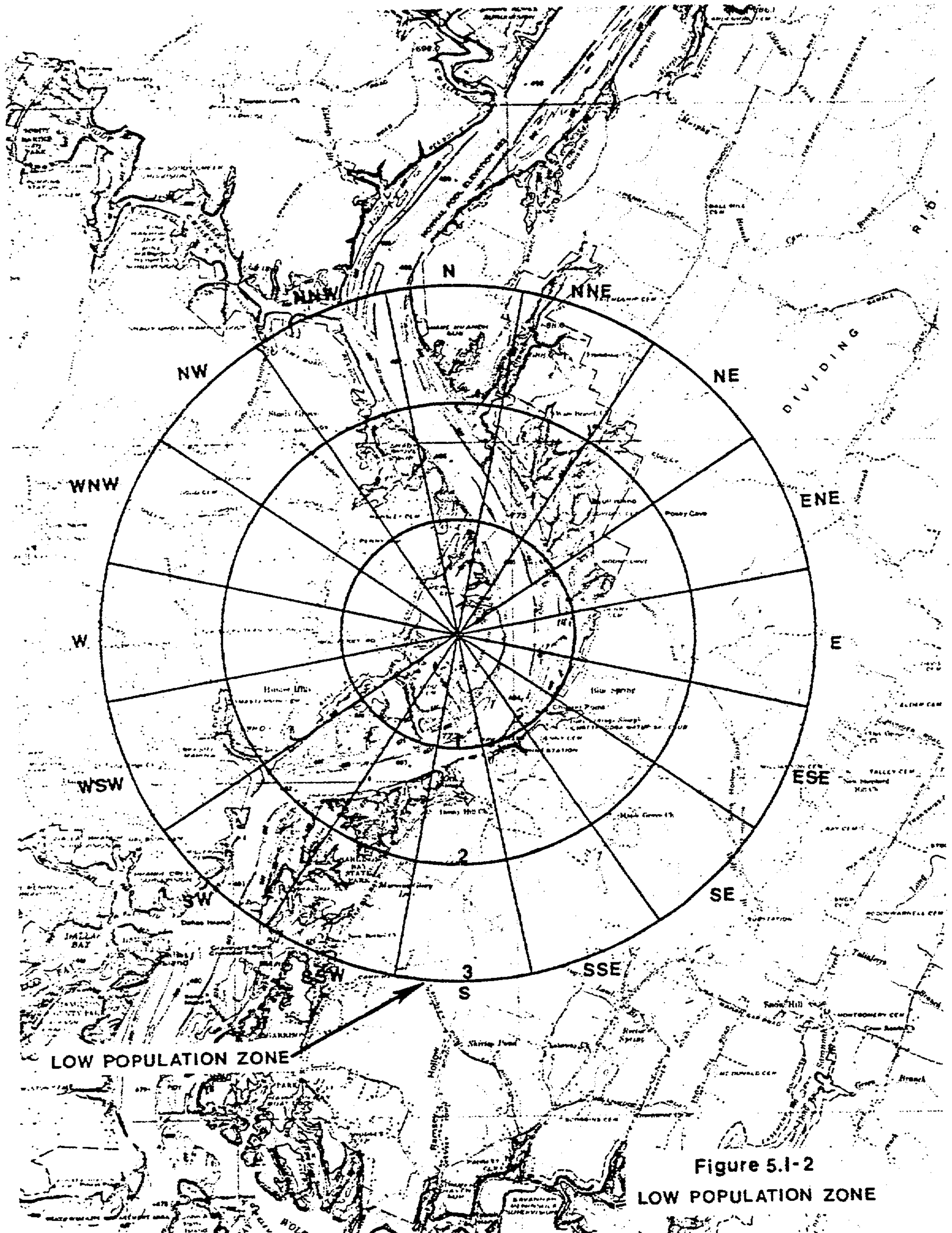

Figure 5.1-2
LOW POPULATION ZONE

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.15 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water, which includes a conservative allowance of 1.42% delta k/k for uncertainties.*

b. A nominal 10.375 inch center-to-center distance between fuel assemblies placed in the storage ranks.

CRITICALITY - NEW FUEL

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21.0 center-to-center distance between new fuel assemblies such that k_{eff} will not exceed 0.98 when fuel having a maximum enrichment of 4.5 weight percent U-235 is in place and optimum achievable moderation is assumed.

DRAINAGE

5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

*For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Director, Nuclear Power Division within 14 days of the violation.

6.8 PROCEDURES & PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Plant Physical Security Plan implementation.

e. Site Radiological Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. Quality Assurance Program for effluent monitoring, using the guidance contained in Regulatory Guide 4.15, December 1977.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PORC and approved by the Plant Superintendent prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PORC and approved by the Plant Superintendent within 14 days of implementation.

6.8.4 Written procedures shall be established, implemented and maintained by the Radiological Hygiene Branch covering the activities below:

a. OFFSITE DOSE CALCULATIONAL MANUAL implementation.

b. Quality Assurance Program and environmental monitoring, using the guidance contained in Regulatory Guide 4.15, December 1977.

c. Surveillance requirements and environmental monitoring requirements shown in Table 6.1-1.

6.8.5 The following programs shall be established, implemented, and maintained:

a. <u>Primary Coolant Sources Outside Containment</u>

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the charging system, safety injection system, residual heat removal system, chemical and volume control system, containment spray system, iodine cleanup system, and hydrogen recombiner system. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. <u>In-Plant Radiation Monitoring</u>

A program which will ensure the capability to accurately determine the airborne iodine concentrations in vital areas under accident conditions. This program shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analysis equipment.

c. <u>Secondary Water Chemistry</u>

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points,

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. DPR-79

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

   A. The applications 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 10, 1981 (two letters), March 1, March 9, and April 7, 1982, comply 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 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 attachments 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. 4, are hereby incorporated into the license.

OFFICE							
SURNAME							
DATE							

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

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

Attachment:
Appendix A Technical
Specification Changes

Date of Issuance: May 4, 1982

OFFICE	LA:DL:LB #4	DL:LB #4	OELD	DL:LB #4	AD:L/DL		
SURNAME	MDuncan/hmc	CStahle	[illegible]	EAdensam	RTedesco		
DATE	4/[illegible]/82	4/[illegible]/82	4/16/82	4/30/82	5/03/82		

ATTACHMENT TO LICENSE AMENDMENT NO. 4

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 area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Overleaf Page	Amended Page
2-8	2-7
3/4 3-76	3/4 3-75
3/4 3-81	3/4 3-82
	3/4 6-19
	3/4 6-20
3/4 6-30	3/4 6-29
3/4 7-43	3/4 7-44
3/4 9-8	3/4 9-7
3/4 12-9	3/4 12-10
5-6	5-5
6-17	6-18
6-20	6-19
5-3	5-4

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 Trip	≤ 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 ≤ 1.14

K_2 = 0.009

TABLE 4.3-8 (Continued)

TABLE NOTATION

* During liquid additions to the tank.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

TABLE 4.3-9 (Continued)

TABLE NOTATION

* At all times.

** During waste gas disposal system operation.

*** During shield building exhaust system operation.

**** During waste gas releases.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

TABLE 3.6-2

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
A. PHASE "A" ISOLATION			
1.	FCV-1-7	SG Blow Dn	10
2.	FCV-1-14	SG Blow Dn	10
3.	FCV-1-25	SG Blow Dn	10
4.	FCV-1-32	SG Blow Dn	10
5.	FCV-1-181	SG Blow Dn	15
6.	FCV-1-182	SG Blow Dn	15
7.	FCV-1-183	SG Blow Dn	15
8.	FCV-1-184	SG Blow Dn	15
9.	FCV-31C-222	CW-Inst Room Clrs	10
10.	FCV-31C-223	CW-Inst Room Clrs	10
11.	FCV-31C-224	CW-Inst Room Clrs	10
12.	FCV-31C-225	CW-Inst Room Clrs	10
13.	FCV-31C-229	CW-Inst Room Clrs	10
14.	FCV-31C-230	CW-Inst Room Clrs	10
15.	FCV-31C-231	CW-Inst Room Clrs	10
16.	FCV-31C-232	CW-Inst Room Clrs	10
17.	FCV-43-22	Sample RC Outlet Hdrs	10
18.	FCV-43-23	Sample RC Outlet Hdrs	10
19.	FCV-43-55	SG Blow Dn Sample Line	10
20.	FCV-43-58	SG Blow Dn Sample Line	10
21.	FCV-43-61	SG Blow Dn Sample Line	10
22.	FCV-43-64	SG Blow Dn Sample Line	10
23.	FCV-61-96	Gylcol Inlet to Floor Cooler	30
24.	FCV-61-97	Gylcol Inlet to Floor Cooler	30
25.	FCV-61-110	Gylcol Outlet to Floor Cooler	30
26.	FCV-61-122	Gylcol Outlet to Floor Cooler	30
27.	FCV-61-191	Ice Condenser - Gylcol In	30
28.	FCV-61-192	Ice Condenser - Gylcol In	30
29.	FCV-61-193	Ice Condenser - Gylcol Out	30
30.	FCV-61-194	Ice Condenser - Gylcol Out	30
31.	FCV-62-61	RCP Seals	10

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
A.	PHASE "A" ISOLATION (Cont.)		
32.	FCV-62-63	RCP Seals	10
33.	FCV-62-72	Letdown Line	10
34.	FCV-62-73	Letdown Line	10
35.	FCV-62-74	Letdown Line	10
36.	FCV-62-77	Letdown Line	10
37.	FCV-63-23	Accum to Hold Up Tank	10
38.	FCV-63-64	WDS N_2 to Accum	10
39.	FCV-63-71	Accum to Hold Up Tank	10
40.	FCV-63-84	Accum to Hold Up Tank	10
41.	FCV-68-305	WDS N_2 to PRT	10
42.	FCV-68-307	PRT to Gas Analyzer	10
43.	FCV-68-308	PRT to Gas Analyzer	10
44.	FCV-70-85	CCS from Excess Lt Dn Hx	10
45.	FCV-70-143	CCS to Excess Lt Dn Hx	60
46.	FCV-77-9	RCDT Pump Disch	10
47.	FCV-77-10	RCDT Pump Disch	10
48.	FCV-77-16	RCDT to Gas Analyzer	10
49.	FCV-77-17	RCDT to Gas Analyzer	10
50.	FCV-77-18	RCDT and PRT to V H	10
51.	FCV-77-19	RCDT and PRT to V H	10
52.	FCV-77-20	N_2 to RCDT	10
53.	FCV-77-127	Floor Sump Pump Disch	10
54.	FCV-77-128	Floor Sump Pump Disch	10
55.	FCV-81-12	Primary Water Makeup	10
56.	FCV-87-7	UHI Test Line	10
57.	FCV-87-8	UHI Test Line	10
58.	FCV-87-9	UHI Test Line	10
59.	FCV-87-10	UHI Test Line	10
60.	FCV-87-11	UHI Test Line	10
61.	FCV-26-240	Fire Protection Isol.	20
62.	FCV-26-243	Fire Protection Isol.	20

CONTAINMENT SYSTEMS

ICE BED TEMPERATURE MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.2 The ice bed temperature monitoring system shall be OPERABLE with at least 2 OPERABLE RTD channels in the ice bed at each of 3 basic elevations (10'6", 30'9", and 55' above the floor of the ice condenser) for each one third of the ice condenser.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the ice bed temperature indication not available in the main control room, determine the ice bed temperature at the local ice condenser temperature monitoring panel every 12 hours.

a. With the ice bed temperature monitoring system inoperable, and unable to determine ice bed temperature at the local panel, POWER OPERATION may continue for up to 30 days provided:

   1. The ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed;

   2. The last recorded mean ice bed temperature was less than or equal to 20°F and steady; and

   3. The ice condenser cooling system is OPERABLE with at least:

      a) 21 OPERABLE air handling units,

      b) 2 OPERABLE glycol circulating pumps, and

      c) 3 OPERABLE refrigerant units;

   Otherwise, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the ice bed temperature monitoring system inoperable and unable to determine the ice bed temperatures by alternate means with the ice condenser cooling system not satisfying the minimum components OPERABILITY requirements of a.3 above, POWER OPERATION may continue for up to 6 days provided the ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed and the last recorded mean ice bed temperature was less than or equal to 15°F and steady; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 The ice bed temperature monitoring system shall be determined OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours.

ACTION: (Continued)

c) In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.11.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by starting each electric motor driven pump and operating it for at least 15 minutes on recirculation flow.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

*c. At least once per 6 months by performance of a system flush.

d. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

e. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that each automatic valve in the flow path actuates to its correct position,

2. Verifying that each pump develops at least 1174 gpm at a system head of 312 feet,

3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

4. Verifying that the No. 1 fire pump starts to maintain the fire suppression water system pressure greater than or equal to 125 psig, and that the No. 2 fire pump starts automatically within 10 ± 2 seconds if the fire suppression water system is not maintained at greater than or equal to 125 psig by the No. 1 pump.

f. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

*Note: These flushes should coincide with the chlorination of the raw service and fire suppression water system. These flushes should be run, one between April 1 and June 30, and the other between September 1 and November 15.

Within the prescribed spring and fall test period, deviation from the six-month performance frequency is authorized.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

   1. A minimum capacity of 2750 pounds, and

   2. An overload cut off limit less than or equal to 2700 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

   1. A minimum capacity of 610 pounds, and

   2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 2750 pounds and demonstrating an automatic electrical load cut off when the crane load exceeds 2700 pounds, and a mechanical load cutoff when the crane load exceeds 2800 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 610 pounds.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of three miles.)

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the maximum value currently being calculated in Specification 4.11.2.3.1, in lieu of any other report required by Specification 6.9.1., prepare and submit to the Commission within 30 days after the updated calculation, pursuant to Specification 6.9.2, a Special Report which identifies the new location(s).

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than the highest calculated dose or dose commitment at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days after the updated calculation, pursuant to Specification 6.9.2, a Special Report which identifies the new location. The new location shall be added to the radiological environmental monitoring program within 30 days, if the owner consents. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per calendar year between the dates of April 1 and October 1 using the following techniques:

1. Within a 2-mile radius from the plant or within the 15-mrem/year isodose line, whichever is larger, enumeration by a door-to-door or equivalent counting technique.

2. With a 5-mile radius from the plant, enumeration by using appropriate techniques such as door-to-door survey, mail survey, telephone survey, serial survey, or information from local agricultural authorities or other reliable sources.

*Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

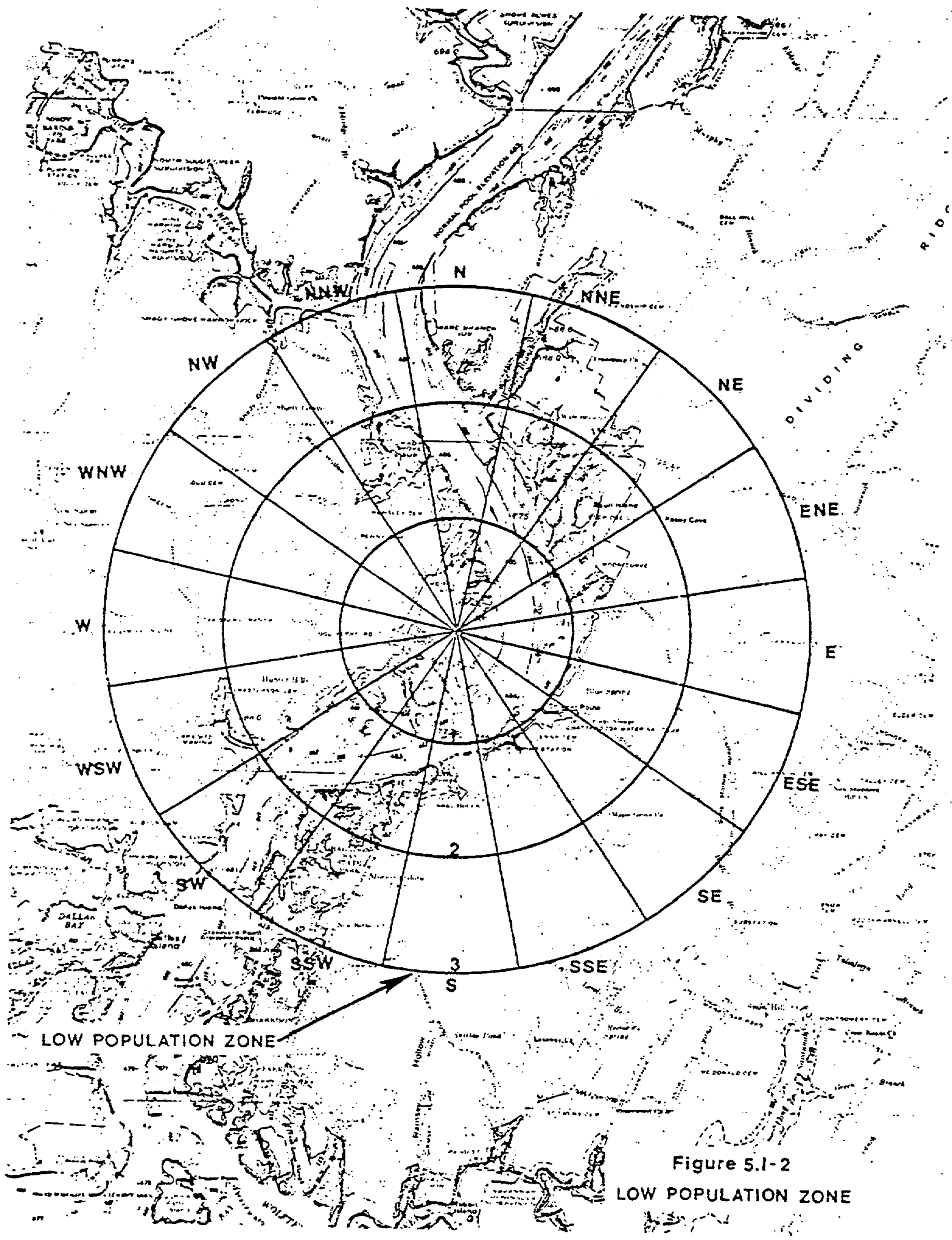

Figure 5.1-2
LOW POPULATION ZONE

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.15 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water, which includes a conservative allowance of 1.42% delta k/k for uncertainties.*

b. A nominal 10.375 inch center-to-center distance between fuel assemblies placed in the storage racks.

CRITICALITY - NEW FUEL

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21.0 inch center-to-center distance between new fuel assemblies such that k_{eff} will not exceed 0.98 when fuel having a maximum enrichment of 4.5 weight percent U-235 is in place and optimum achievable moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

*For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Director, Nuclear Power Division within 14 days of the violation.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Plant Physical Security Plan implementation.

e. Site Radiological Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. Quality Assurance Program for effluent monitoring, using the guidance contained in Regulatory Guide 4.15, December 1977.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PORC and approved by the Plant Superintendent prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PORC and approved by the Plant Superintendent within 14 days of implementation.

6.8.4 Written procedures shall be established, implemented and maintained by the Radiological Hygiene Branch covering the activities below:

a. OFFSITE DOSE CALCULATIONAL MANUAL implementation.

b. Quality Assurance Program and environmental monitoring, using the guidance contained in Regulatory Guide 4.15, December 1977.

c. Surveillance requirements and environmental monitoring requirements shown in Table 6.1-1.

6.8.5 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the safety injection system, residual heat removal system, chemical and volume control system, containment spray system, and RCS sampling system. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentrations in vital areas under accident conditions. This program shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points,

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE DPR-77

AND AMENDMENT NO. 4 TO FACILITY OPERATING LICENSE DPR-79

TENNESSEE VALLEY AUTHORITY

INTRODUCTION

This SER covers the following TVA requests for amendments to the Technical Specifications:

A. Proposed changes dated 12/10/81 on Fire Hose Hydrostatic Test Pressure, Ice Bed Determination and Surveillance, Maximum Isolation Valve Times, Vital Battery Surveillance;

B. Proposed changes, dated 12/10/81 on a table correction, QA Monitoring of Plant Effluents, Land Use Census, and System Flushing;

C. Proposed changes, dated 3/9/82, on Downscale Failure Alarms;

D. Proposed changes, dated 4/7/82, on Visual Inspection Schedule on Snubbers; and

E. Proposed changes, dated 3/1/82, on Maximum Enrichment for Reload Fuel.

EVALUATION

ITEM A

The proposed changes related to vital battery surveillance and the fire hose test pressure requirements were made in Amendment 12 to the Technical Specifications of Unit 1. The original Technical Specifications of Unit 2 incorporated the proposed changes.

The alternate measurement for the determination of ice bed temperature (if the control room monitor fails) was considered acceptable since the use of a digital voltmeter is more accurate than the Control Room monitor.

The proposed change from 10 seconds to 30 seconds for the eight glycol valves to close is acceptable because the response time conforms with the safety analysis on isolating the containment reported in the SER, NUREG-0011. The glycol valves are part of the refrigeration system for the ice bed, and they do not provide a direct leakage path to the environs. For such valves, closure times of less than 60 seconds are acceptable (See SRP section 6.2.4 NUREG-0800).

ITEM B

Table 3.6-2 for both units is revised to add a containment isolation valve to the list which was inadvertently left off by the licensee. This valve for the units has been tested even though it was not on the table in the Technical Specifications.

The change clarifies the responsibilities for implementing the QA program for effluent monitoring at the site. This is an administrative change.

The Technical Specifications are revised to define more specifically the survey of land use census that is performed. This revision is for clarification purposes. No actual changes are being made in the performance of the census.

The change provides greater coordination between the surveillance requirements for the fire suppression water system flush with the chlorination of the raw service water and the fire suppression systems. Chlorination is a seasonal process and coordinating this with the fire suppression system flush promotes greater efficiency without degradation of the systems.

ITEM C

Downscale failure alarms are now installed for units 1 & 2. This change is made to Tables 4.3-8 and 4.3-9 to reflect that such an installation has occurred. This revision is adminstrative in nature.

ITEM D

In order to preclude a forced outage of Unit 1, the requirement for visual inspection of the snubbers was changed to state that such an inspection would be performed at an outage of sufficient duration (approximately 72 hours in Mode 5) but no later than 18 months ±25% from the first inspection. The completion of the second inspection period is changed from 12 months to 18 months which is judged to be acceptable by the staff because the majority of the snubbers have been inspected and only a few remain to be inspected in Mode 5. These are in areas that are impossible to inspect during operations.

ITEM E

Tennessee Valley Authority has submitted a proposed amendment requesting changes in the maximum fuel enrichment for reload fuel from 3.5 to 4.0 w/o U-235 and changes in the maximum fuel enrichment for fuel in the new fuel pit storage racks from 3.5 to 4.5 w/o U-235 (Letter from L. M. Mills to H. R. Denton, TVA-SNP-TS-26, dated March 1, 1982) for the Sequoyah Units 1 and 2.

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The criticality aspects of the new and spent fuel storage racks have been analyzed by TVA using the KENO-IV Monte Carlo code with cross sections generated by the AMPX code. These codes have been benchmarked against a set of 27 critical experiments ranging from water moderated, oxide fuel arrays separated by various materials (Boral, steel and water) to dry, harder spectrum uranium metal cylinder arrays with various interspersed materials (Plexiglass, steel and air). The average K-effective calculated for these benchmarks is 0.9998 which demonstrates that there is no bias associated with the analytical method over the wide range of experimental conditions analyzed.

New Fuel

Criticality of fuel assemblies in the new fuel storage rack is prevented by restricting the minimum separation between assemblies to 21 inches center-to-center. The analysis by TVA assumed the highest allowable enrichment (4.5 w/o U-235) without any control rods or any noncontained burnable poison at its most reactive point in life (no depletion or fission product buildup). The fuel racks are assumed to be infinite in lateral and axial extent. Although the new fuel storage pit is designed to be dry, the maximum K-effective must be determined by considering credible accidents. The most limiting credible accident has been found to be the introduction of moderator into the new fuel storage pit. The optimum achievable moderation occurs with full density water (1.0 g/cc) and the calculations were performed at this condition.

Under these assumptions the nominal K-effective for the new fuel storage racks in their design configuration is 0.9189 as determined by the KENO-IV code. A mechanical bias of 0.001 is added to this value to account for the fact that mechanical tolerances can result in spacings between assemblies less than nominal. A total uncertainty at a 95/95 probability/confidence level which includes the statistical uncertainty associated with the nominal case KENO K-effective and the statistical uncertainty due to the method is also applied. The resulting K-effective, including all of these biases and uncertainties, is 0.9343, well below the acceptance criterion of 0.98 for new fuel with optimum moderation, and the acceptance criterion of 0.95 for new fuel fully flooded with nonborated water.

We conclude that the modification to the Sequoyah Technical Specification 5.6.1.2 increasing the maximum fuel enrichment for fuel in the new fuel pit storage racks for Units 1 and 2 from 3.5 w/o U-235 to 4.5 w/o U-235 is acceptable. The maximum enrichment of fuel allowed in the core is, however, 4.0 w/o U-235. (T.S. 5.3.1)

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Spent Fuel

Criticality of fuel assemblies in the spent fuel storage rack is prevented by restricting the minimum separation between assemblies to 10.375 inches center-to-center and inserting neutron poison (Boral) between assemblies. The analysis assumed that highest allowable enrichment (4.0 w/o U-235) without any control rods or any noncontained burnable poison at its most reactive point in life (no depletion or fission product buildup). The fuel racks are assumed to be infinite in lateral and axial extent. The moderator is pure water at a density of 1.0 g/cc. No dissolved boron is included in the water. Credit is taken for the neutron absorption in full length structural materials and in solid materials added specifically for neutron absorption. The minimum boron loading of 0.0232 gm (B10)/cm^3 is assumed in the poison plates.

Under these assumptions the nominal K-effective for the spent fuel storage racks in their design configuration is 0.92304 as determined by KENO-IV. No mechanical bias was included since studies indicated that the nominal case assumptions result in the worse case. A bias of 0.0025 to account for B_4C particle self-shielding is included. A total uncertainty at a 95/95 probability/confidence level which includes the statistical uncertainty associated with mechanical tolerances, uncertainty in the nominal case K-effective, and uncertainty in the method bias, is also applied. The resulting K-effective, including all of these biases and uncertainties, is 0.9397, below the acceptance criterion of 0.95.

The effect of credible accidents has been considered and the most consequential one is dropping of a single fuel assembly outside the rack between the periphery of the storage racks and the side walls of the pool. For accident conditions, TVA has applied the double contingency principle of ANS N16.1-1975 which states that it shall require two unlikely, independent, concurrent events to produce a criticality accident. This has been accepted by the staff. Therefore, the presence of soluble boron in the storage pool water can be assumed for accident conditions such as the fuel assembly drop. Since 2000 ppm of boron in the pool water will decrease reactivity by more than 30%, any postulated accidental reactivity increase will be much less than the negative worth of the dissolved boron.

In a phone conversation on March 12, 1982, between TVA and NRC, a modification to Technical Specification 5.6.1.1 was agreed upon. The reference to the double contingency principle was removed from the Technical Specification itself and inserted as a footnote.

We conclude that the modification to Technical Specification 5.6.1.1, allowing the assumption of the double contingency principle (as a footnote) in determining the criticality of spent fuel storage racks is acceptable.

Our evaluation and approval of these Technical Specification modifications is based on PWR fuel pins and fuel assemblies similar in design to the Westinghouse 17x17 fuel presently installed in the Sequoyah Units 1 and 2. Fuel designs differing from this may require a reevaluation even though the U-235 enrichment and fuel assembly spacing specifications remain the same.

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 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.

Dated: May 4, 1982

Principal Contributors: Carl Stahle, Licensing Branch No. 4, DOL
Laurence Kopp, Core Performance Branch, DSI
Norman Wagner, Auxiliary Systems Branch, DSI

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-327 AND 50-328

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS

FACILITY OPERATING LICENSE NOS. DPR-77 AND DPR-79

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 13 to Facility Operating License No. DPR-77 and Amendment No. 4 to Facility Operating License DPR-79, issued to Tennessee Valley Authority (licensee) for the Sequoyah Plant, Units 1 and 2 (the facilities) located in Hamilton County, Tennessee. These amendments change various sections of the Technical Specifications related to fire hose test pressure requirements, vital battery surveillance, ice bed temperature determination, isolation valve closing times, downscale failure alarms, visual inspection schedule on snubbers, maximum enrichment for reload fuel, quality assurance monitoring of plant effluents, land use census, and system flushing. The amendments are effective as of their dates of issuance.

The applications for the amendments comply 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 amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) environmental impact statements, or negative declarations and environmental impact appraisals need not be prepared in connection with issuance of these amendments.

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For further details with respect to this action, see (1) Tennessee Valley Authority letters dated December 10, 1981 (two letters), March 1, March 9, and April 6, 1982, (2) Amendment No. 13 to Facility Operating License No. DPR-77 with Appendix A Technical Specification page changes; (3) Amendment No. 4 to Facility Operating License No. DPR-79 with Appendix A Technical Specification page changes; and (4) 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. 13 and Amendment No. 4 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 4th day of May 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

S/

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

OFFICE	LA:DL:LB #4	DL:LB #4	OELD	DL:LB #4
SURNAME	MDuncan/hmc	CStahle		EAdensam
DATE	4/14/82	4/16/82	4/ /82	4/30/82

April 15, 1982

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D. Eisenhut
OELD

Docket Nos: 50-327
and 50-328

MEMORANDUM FOR: Robert L. Tedesco, Assistant Director for Licensing
Division of Licensing, NRR

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

FROM: Carl Stahle, Project Manager
Licensing Branch No. 4
Division of Licensing, NRR

SUBJECT: ISSUANCE OF AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE DPR-77 AND AMENDMENT NO. 4 TO FACILITY OPERATING LICENSE DPR-79, SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

Regarding the issuance of subject amendments, there is no known public correspondence or irreversible impact associated with this subject.

S/

Carl Stahle, Project Manager
Licensing Branch No. 4
Division of Licensing
Office of Nuclear Reactor Regulation

OFFICE	LA:DL:LB #4	DL:LB #4	DL:LB #4				
SURNAME	MDuncan/hmc	CStahle	EAdensam				
DATE	4/14/82	4/15	4/30/82				