March 31, 1986

Docket Nos. 50-259/260/296

Mr. S. A. White
Manager of Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37401

Dear Mr. White:

The Commission has issued the enclosed Amendment Nos. 128 , 123 and 99 to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3. These amendments are in response to your application dated October 1, 1985 (TVA BFNP TS-213).

The amendments change the Technical Specifications to correct inconsistencies and typographical errors, and to add new surveillance requirements.

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

Sincerely,

Richard J. Clark, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 128 to
   License No. DPR-33
2. Amendment No. 123 to
   License No. DPR-52
3. Amendment No. 99 to
   License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION	JPartlow		
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Local PDR	TBarnhart (4cys for each docket)		LFMB
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*Please see previous concurrence page

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Mr. S. A. White
Tennessee Valley Authority

Browns Ferry Nuclear Plant
Units 1, 2, and 3

cc:

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 128
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 1, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-33 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 128, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

V Rooney for

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 31, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 128

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages.

v

9

108

157

179

185

227

262

2. The marginal lines on these pages denote the area being changed.

Section		Page No.
6.2	Review and Audit	333
6.3	Procedures	338
6.4	Actions to be Taken in the Event of a Reportable Occurrence in Plant Operation .	346
6.5	Actions to be taken in the Event a Safety Limit is Exceeded	346
6.6	Station Operating Records	346
6.7	Reporting Requirements	349
6.8	Minimum Plant Staffing	358

Amendment No. 92, 128,

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1 FUEL CLADDING INTEGRITY	2.1 FUEL CLADDING INTEGRITY

b. For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

(Note: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are LHGR<13.4 kw/ft for 8x8, 8x8R, and P8x8R fuel, MCPR limits of Spec 3.5.k. If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within prescribed limits. Surveillance requirements for APRM scram setpoint are given in specification 4.5.L.

c. The APRM Rod block trip setting shall be:

$S_{RB} \leq (0.66W + 42\%)$

where:

S_{RB} = Rod block setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

Amendment No. ~~70~~, ~~85~~, 128,

Table 4.2.J

Seismic Monitoring Instrument Surveillance Requirements

INSTRUMENT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
TRIAXIAL TIME HISTORY ACCELOGRAPHS			
a. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly*	6 months	NA
b. Unit 1 reactor bldg. floor slab (El. 621.25)	Monthly*	6 months	NA
c. Diesel-generator bldg base slab (El. 565.5)	Monthly*	6 months	NA
BIAXIAL SEISMIC SWITCHES			
a. Unit 1 reactor bldg. base slab	Monthly*	6 months	once/operating cycle
b. Unit 1 reactor bldg. base slab	Monthly*	6 months	once/operating cycle
c. Unit 1 reactor bldg. base slab	Monthly*	6 months	once/operating cycle
TRIAXIAL PEAK ACCELOGRAPHS			
a. U-1 RBCCW, 10" pipe (El. 625.75)	NA	12 months	N/A
b. U-1 RHRSW, 16" pipe (El. 580.0)	NA	12 months	N/A
c. U-1 core spray system, 14" pipe (El. 544.0)	NA	12 months	N/A

*Except seismic switches

3.5.F Reactor Core Isolation Cooling

2. If the RCICS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is operable during such time.

3. If specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurizeed to less than 122 psig within 24 hours.

G. Automatic Depressurization System (ADS)

1. Four of the six valves of the Automatic Depressurization System shall be operable:

   (1) prior to a startup from a Cold Condition, or,

   (2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except as specified in 3.5.G.2 and 3.5.G.3 below.

2. If three of the six ADS valves are known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is operable. (Note that the pressure relief function of these valves is assured by section 3.6.D of these specifications and that this specification only applies to the ADS function.) If more than three of the six ADS valves are known to be incapable of automatic operation, an immediate orderly shutdown shall be initiated, with the reactor in a hot shutdown condition in 6 hours and in a cold shutdown condition in the following 18 hours.

4.5.F Reactor Core Isolation Cooling

2. When it is determined that the RCICS in inoperable, the HPCIS shall be demonstrated to be operable immediately.

G. Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

   a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage. Manual surveillance of the relief valves is covered, in 4.6.D.2.

2. When it is determined that three of the six ADS valves are incapable of automatic operation, the HPCIS shall be demonstrated to be operable immediately and daily thereafter as long as Specification 3.5.G.2 applies.

3.6 PRIMARY SYSTEM BOUNDARY

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μc/gm of dose equivalent* I-131.

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 μCi/gm whenever the reactor is critical. The reactor shall not be operated more than 5 percent of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 μCi/gm, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

* That concentration of I-131 which alone would produce the same thyroid dose as the quantity of total iodines actually present.

4.6 PRIMARY SYSTEM BOUNDARY

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

   a. During startup
   b. Following a significant power change**
   c. Following an increase in the equilibrium off-gas level exceeding 10,000 μCi/sec (at the steam jet air ejector) within a 48 hour period.
   d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 μCi/gm) is established. However, at least 3 consecutive samples ahall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration. If the total iodine activity of the sample is below 0.32 uci/gm, an isotopic analysis to determine equivalent I-131 is not required.

** For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

3.6 PRIMARY SYSTEM BOUNDARY

H. Seismic Restraints, Supports, and Snubbers

1. During all modes of operation except Cold Shutdown and Refuel, and seismic restraints, supports, and snubbers shall be operable except as noted in 3.6.H.2 and 3.6.H.3 below. All safety-related snubbers are listed in Surveillance Instruction BF SI 4.6.H-1 & -2.

2. With one or more seismic restraint , support, or snubber inoperable; within 72 hours replace or restore the inoperable seismic restraint(s), support(s), or snubber(s), to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate LIMITING CONDITION statement for that system.

3. If a seismic restraint, support, or snubber (SRSS) is determined to be inoperable while the reactor is in the shutdown or refuel mode, that SRSS shall be made operable or replaced prior to reactor startup. If the inoperable SRSS is attached to a system that is required OPERABLE during the shutdown or refuel mode, the appropriate LIMITING CONDITIONS statement for that system shall be followed.

4.6 PRIMARY SYSTEM BOUNDARY

H. Seismic Restraints, Supports, and Snubbers

The surveillance requirements of paragraph 4.6.G are the only requirements that apply to any seismic restraint or support other than snubbers.

Each safety-related snubber shall be demonstrated OPERABLE BY performance of the following augumented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Surveillance Instructions BF SI 4.6.H-1 and -2.

1. 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 groups based on design, environment, or other features which may be expected to affect the operability of the snubbers within the group. Each group may be inspected independently in accordance with 4.6.H.2 through 4.6.H.9.

2. Visual Inspection, Schedule, and Lot Size

The first inservice visual inspection of snubbers not previously included in these technical specifications and whose visual inspection has not been performed and documented previously, shall be performed within six months for accessible snubbers and before resuming power after the first refueling outage

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

a. Minimum water level = -6.25" (differential pressure control >0 psid)

-7.25" (0 psid differential pressure control)

b. Maximum water level = -1"

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

TABLE 3.7.E

PRIMARY CONTAINMENT ISOLATION VALVES WHICH TERMINATE BELOW THE SUPPRESSION POOL WATER LEVEL

Valve	Valve Identification
12-738	Auxiliary Boiler to RCIC
12-741	Auxiliary Boiler to RCIC
43-28A	RHR Suppression Chamber Sample Lines
43-28B	RHR Suppression Chamber Sample Lines
43-29A	RHR Suppression Chamber Sample Lines
43-29B	RHR Suppression Chamber Sample Lines
2-1143	Demineralized Water
71-14	RCIC Turbine Exhaust
71-32	RCIC Vacuum Pump Discharge
71-580	RCIC Turbine Exhaust
71-592	RCIC Vacuum Pump Discharge
73-23	HPCI Turbine Exhaust
73-24	HPCI Turbine Exhaust Drain
73-603	HPCI Turbine Exhaust
73-609	HPCI Exhaust Drain
74-722	RHR
75-57	Suppression Chamber Drain
75-58	Suppression Chamber Drain

Amendment No. ~~86~~, 128

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 123
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 1, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 123, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

P Rooney for

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 31, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 123

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages.

v

108

157

179

185

227

253a

260

262

2. The marginal lines on these pages denote the area being changed.

Section		Page No.
6.3	Procedures	338
6.4	Actions to be Taken in the Event of a Reportable Occurrence in Plant Operation	346
6.5	Actions to be Taken in the Event a Safety Limit is Exceeded	346
6.6	Station Operating Records	346
6.7	Reporting Requirements	349
6.8	Minimum Plant Staffing	358

Amendment No. ~~85~~, 123,

Table 4.2.J

Seismic Monitoring Instrument Surveillance Requirements

INSTRUMENT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
TRIAXIAL TIME HISTORY ACCELOGRAPHS			
a. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly*	6 months	NA
b. Unit 1 reactor bldg. floor slab (El. 621.25)	Monthly*	6 months	NA
c. Diesel-generator bldg base slab (El. 565.5)	Monthly*	6 months	NA
BIAXIAL SEISMIC SWITCHES			
a. Unit 1 reactor bldg. base slab	Monthly*	6 months	once/operating cycle
b. Unit 1 reactor bldg. base slab	Monthly*	6 months	once/operating cycle
c. Unit 1 reactor bldg. base slab	Monthly*	6 months	once/operating cycle
TRIAXIAL PEAK ACCELOGRAPHS			
a. U-1 RBCCW, 10" pipe (El. 625.75)	NA	12 months	N/A
b. U-1 RHRSW, 16" pipe (El. 580.0)	NA	12 months	N/A
c. U-1 core spray system, 14" pipe (El. 544.0)	NA	12 months	N/A

*Except seismic switches

3.5.F Reactor Core Isolation Cooling

2. If the RCICS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is operable during such time.

3. If specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurizeed to less than 122 psig within 24 hours.

G. Automatic Depressurization System (ADS)

1. Four of the six valves of the Automatic Depressurization System shall be operable:

(1) prior to a startup from a Cold Condition, or,

(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except as specified in 3.5.G.2 and 3.5.G.3 below.

2. If three of the six ADS valves are known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is operable. (Note that the pressure relief function of these valves is assured by section 3.6.D of these specifications and that this specification only applies to the ADS function.) If more than three of the six ADS valves are known to be incapable of automatic operation, an immediate orderly shutdown shall be initiated, with the reactor in a hot shutdown condition in 6 hours and in a cold shutdown condition in the following 18 hours.

4.5.F Reactor Core Isolation Cooling

2. When it is determined that the RCICS in inoperable, the HPCIS shall be demonstrated to be operable immediately.

G. Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage. Manual surveillance of the relief valves is covered, in 4.6.D.2.

2. When it is determined that three of the six ADS valves are incapable of automatic operation, the HPCIS shall be demonstrated to be operable immediately and daily thereafter as long as Specification 3.5.G.2 applies.

3.6 PRIMARY SYSTEM BOUNDARY

6. Whenever the reactor is critical, the limits on activity concentrations in the reactor coolant shall not exceed the equilibrium value of 3.2 μc/gm of dose equivalent* I-131.

This limit may be exceeded following power transients for a maximum of 48 hours. During this activity transient the iodine concentrations shall not exceed 26 μCi/gm whenever the reactor is critical. The reactor shall not be operated more than 5 percent of its yearly power operation under this exception for the equilibrium activity limits. If the iodine concentration in the coolant exceeds 26 μCi/gm, the reactor shall be shut down, and the steam line isolation valves shall be closed immediately.

*That concentration of I-131 which alone would produce the same thyroid dose as the quantity of total iodines actually present.

4.6 PRIMARY SYSTEM BOUNDARY

6. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.6 and one of the following conditions are met:

   a. During startup
   b. Following a significant power change**
   c. Following an increase in the equilibrium off-gas level exceeding 10,000 μCi/sec (at the steam jet air ejector) within a 48 hour period.
   d. Whenever the equilibrium iodine limit specified in 3.6.B.6 is exceeded.

The additional coolant liquid samples shall be taken at 4 hour intervals for 48 hours, or until a stable iodine concentration below the limiting value (3.2 μCi/gm) is established. However, at least 3 consecutive samples ahall be taken in all cases. An isotopic analysis shall be performed for each sample, and quantitative measurements made to determine the dose equivalent I-131 concentration. If the total iodine activity of the sample is below 0.32 μci/gm, an isotopic analysis to determine equivalent I-131 is not required.

**For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

3.6 PRIMARY SYSTEM BOUNDARY

H. Seismic Restraints, Supports, and Snubbers

1. During all modes of operation except Cold Shutdown and Refuel, and seismic restraints, supports, and snubbers shall be operable except as noted in 3.6.H.2 and 3.6.H.3 below. All safety-related snubbers are listed in Surveillance Instruction BF SI 4.6.H.

2. With one or more seismic restraint , support, or snubber inoperable; within 72 hours replace or restore the inoperable seismic restraint(s), support(s), or snubber(s), to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate LIMITING CONDITION statement for that system.

3. If a seismic restraint, support, or snubber (SRSS) is determined to be inoperable while the reactor is in the shutdown or refuel mode, that SRSS shall be made operable or replaced prior to reactor startup. If the inoperable SRSS is attached to a system that is required OPERABLE during the shutdown or refuel mode, the appropriate LIMITING CONDITIONS statement for that system shall be followed.

4.6 PRIMARY SYSTEM BOUNDARY

H. Seismic Restraints, Supports, and Snubbers

The surveillance requirements of paragraph 4.6.G are the only requirements that apply to any seismic restraint or support other than snubbers.

Each safety-related snubber shall be demonstrated OPERABLE BY performance of the following augumented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Surveillance Instructions BF SI 4.6.H-1 and -2.

1. 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 groups based on design, environment, or other features which may be expected to affect the operability of the snubbers within the group. Each group may be inspected independently in accordance with 4.6.H.2 through 4.6.H.9.

2. Visual Inspection, Schedule, and Lot Size

The first inservice visual inspection of snubbers not previously included in these technical specifications and whose visual inspection has not been performed and documented previously, shall be performed within six months for accessible snubbers and before resuming power after the first refueling outage

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

   a. Minimum water level = -6.25" (differential pressure control >0 psid)

   -7.25" (0 psid differential pressure control)

   b. Maximum water level = -1"

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

   a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
6	Drywell ΔP air compressor suction valve (FCV-64-139)		1	10	C	SC
6	Drywell ΔP air compressor discharge valve (FCV-64-140)		1	10	C	SC
6	Drywell CAM suction valves (FCV-90-254A and 254B)		2	10	O	GC
6	Drywell CAM discharge valves (FCV-90-257A and 257B)		2	10	O	GC
6	Drywell CAM suction valve (FCV-90-255)		1	10	O	GC

TABLE 3.7.D (Continued)

Valve	Valve Identification
90-254B	Radiation Monitor Suction
90-255	Radiation Monitor Suction
90-257A	Radiation Monitor Discharge
90-257B	Radiation Monitor Discharge

TABLE 3.7.E

PRIMARY CONTAINMENT ISOLATION VALVES WHICH TERMINATE
BELOW THE SUPPRESSION POOL WATER LEVEL

Valve	Valve Identification
12-738	Auxiliary Boiler to RCIC
12-741	Auxiliary Boiler to RCIC
43-28A	RHR Suppression Chamber Sample Lines
43-28B	RHR Suppression Chamber Sample Lines
43-29A	RHR Suppression Chamber Sample Lines
43-29B	RHR Suppression Chamber Sample Lines
2-1143	Demineralized Water
71-14	RCIC Turbine Exhaust
71-32	RCIC Vacuum Pump Discharge
71-580	RCIC Turbine Exhaust
71-592	RCIC Vacuum Pump Discharge
73-23	HPCI Turbine Exhaust
73-24	HPCI Turbine Exhaust Drain
73-603	HPCI Turbine Exhaust
73-609	HPCI Exhaust Drain
74-722	RHR
75-57	Suppression Chamber Drain
75-58	Suppression Chamber Drain

Amendment No. 85, 123,

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 99
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 1, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-68 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 99 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Rooney for

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 31, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 99

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages.

105

120

161

188

198

231

264A

279

2. The marginal lines on these pages denote the area being changed.

Amendment No. 99,

Table 4.2.J

Seismic Monitoring Instrument Surveillance Requirements

INSTRUMENT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
TRIAXIAL TIME HISTORY ACCELOGRAPHS			
a. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly*	6 months	NA
b. Unit 1 reactor bldg. floor slab (El. 621.25)	Monthly*	6 months	NA
c. Diesel-generator bldg. base slab (El. 565.5)	Monthly*	6 months	NA
BIAXIAL SEISMIC SWITCHES			
a. Unit 1 reactor bldg. base slab	Monthly*	6 months	once/operating cycle
b. Unit 1 reactor bldg. base slab	Monthly*	6 months	once/operating cycle
c. Unit 1 reactor bldg. base slab	Monthly*	6 months	once/operating cycle
TRIAXIAL PEAK ACCELOGRAPHS			
a. U-1 RBCCW, 10" pipe (El. 625.75)	NA	12 months	N/A
b. U-1 RHRSW, 16" pipe (El. 580.0)	NA	12 months	N/A
c. U-1 core spray system, 14" pipe (El. 544.0)	NA	12 months	N/A

*Except seismic switches

3.3 REACTIVITY CONTROL

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

d. Control rods with a failed "Full-in" or "Full-Out" position switch may be bypassed in the Rod Sequence Control System and considered operable if the actual rod position is known. These rods must be moved in sequence to their correct positions (full in on insertion or full out on withdrawal).

4.3 REACTIVITY CONTROL

c. When it is initially determined that a control rod is incapable of normal insertion a test shall be conducted to demonstrate that the cause of the malfunction is not a failure in the control rod drive mechanism. If this can be demonstrated an attempt to fully insert the control rod shall be made. If the control rod cannot be inserted and an investigation has demonstrated that the cause of failure is not a failed control rod drive mechanism collet housing, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined, highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted.

d. The control rod accumulators shall be determined operable at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

G. Automatic Depressurization System (ADS)

1. Four of the six valves of the Automatic Depressurization System shall be operable:

(1) prior to a startup from a Cold Condition, or,

(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except as specified in 3.5.G.2 and 3.5.G.3 below.

2. If three of the six ADS valves are known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is operable. (Note that the pressure relief function of these valves is assured by section 3.6.D of these specifications and that this specification only applies to the ADS function.) If more than three of the six ADS valves are known to be incapable of automatic operation, an immediate orderly shutdown shall be initiated, with the reactor in a hot shutdown condition in 6 hours and in a cold shutdown condition in the following 18 hours.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

G. Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage. Manual surveillance of the relief valves is covered, in 4.6.D.2.

2. When it is determined that three of the six ADS valves are incapable of automatic operation the HPCIS shall be demonstrated to be operable immediately and daily thereafter as long as Specification 3.5.G.2 applies.

3.6 PRIMARY SYSTEM BOUNDARY

3. At steaming rates greater than 100,000 lb/hr, the reactor water quality may exceed specification 3.6.B.2 only for the time limits specified below. Exceeding these time limits of the following maximum quality limits shall be cause for placing the reactor in the cold shutdown condition.

   a. Conductivity time above 2 μmho/cm@25°C - 4 weeks/year. Maximum Limit 10 μmho/cm@25°C

   b. Chloride concentration time above 0.2 ppm - 4 weeks/year. Maximum Limit - 0.5 ppm.

4.6 PRIMARY SYSTEM BOUNDARY

3. Additional coolant samples shall be taken whenever the reactor activity exceeds one percent of the equilibrium concentration specified in 3.6.B.5 and one of the following conditions are met:

   a. During startup

   b. Following a significant power change**

   c. Following an increase in the equilibrium off-gas level exceeding 10,000 uci/sec (at the steam jet air ejector) within a 48 hour period.

   d. Whenever the equilibrium iodine limit specified in 3.6.B.5 is exceeded.

**For the purpose of this section on sampling frequency, a significant power exchange is defined as a change exceeding 15% of rated power in less than 1 hour.

3.6 PRIMARY SYSTEM BOUNDARY

H. Seismic Restraints, Supports, and Snubbers

1 During all modes of operation except Cold Shutdown and Refuel, and seismic restraints, supports, and snubbers shall be operable except as noted in 3.6.H.2 and 3.6.H.3 below. All safety-related snubbers are listed in Surveillance Instruction BF SI 4.6.H.

2. With one or more seismic restraint , support, or snubber inoperable; within 72 hours replace or restore the inoperable seismic restraint(s), support(s), or snubber(s), to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate LIMITING CONDITION statement for that system.

3. If a seismic restraint, support, or snubber (SRSS) is determined to be inoperable while the reactor is in the shutdown or refuel mode, that SRSS shall be made operable or replaced prior to reactor startup. If the inoperable SRSS is attached to a system that is required OPERABLE during the shutdown or refuel mode, the appropriate LIMITING CONDITIONS statement for that system shall be followed.

4.6 PRIMARY SYSTEM BOUNDARY

H. Seismic Restraints, Supports, and Snubbers

The surveillance requirements of paragraph 4.6.G are the only requirements that apply to any seismic restraint or support other than snubbers.

Each safety-related snubber shall be demonstrated OPERABLE BY performance of the following augumented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Surveillance Instructions BF SI 4.6.H-1 and -2.

1. 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 groups based on design, environment, or other features which may be expected to affect the operability of the snubbers within the group. Each group may be inspected independently in accordance with 4.6.H.2 through 4.6.H.9.

2. Visual Inspection, Schedule, and Lot Size

The first inservice visual inspection of snubbers not previously included in these technical specifications and whose visual inspection has not been performed and documented previously, shall be performed within six months for accessible snubbers and before resuming power after the first refueling outage

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification

A. Primary Containment

1. At any time that the irradiated fuel is in the reactor vessel, and the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water level and temperature shall be maintained within the following limits.

   a. Minimum water level = -6.25" (differential pressure control >0 psid)

      -7.25" (0 psid differential pressure control)

   b. Maximum water level = -1"

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

   a. The suppression chamber water level be checked once per day. Whenever heat is added to the suppression pool by testing of the ECCS or relief valves the pool temperature shall be continually monitored and shall be observed and logged every 5 minutes until the heat addition is terminated.

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (Sec.)	Normal Position	Action on Initiating Signal
6	Torus Oxygen Sample Line Valves-Analyzer B (FSV-76-63, 64)		2	NA	Note 1	SC
6	Drywell Hydrogen Sample Line Valves-Analyzer B (FSV-76-59, 60)		2	NA	Note 1	SC
6	Drywell Oxygen Sample Line Valves-Analyzer B (FSV-76-61, 62)		2	NA	Note 1	SC
6	Sample Return Valves-Analyzer B (FSV-76-67, 68)		2	NA	0	GC
7	RCIC Steamline Drain (FSV-71-6A, 6B)		2	5	0	GC
7	RCIC Condensate Pump Drain (FCV-71-7A, 7B)		2	5	C	SC
7	HPCI Hotwell pump discharge isolation valves (FCV-73-17A, 17B)		2	5	C	SC
7	HPCI steamline drain (FCV-73-6A, 6B)		2	5	0	GC
8	TIP Guide Tubes (5)		1 per guide tube	NA	C	GC

NOTE: 1: Analyzers are such that one is sampling drywell hydrogen and oxygen (valves from drywell open - valves from torus closed) while the other is sampling torus hydrogen and oxygen (valves from torus open - valves from drywell closed)

TABLE 3.7.E

PRIMARY CONTAINMENT ISOLATION VALVES WHICH TERMINATE
BELOW THE SUPPRESSION POOL WATER LEVEL

Valve	Valve Identification
12-738	Auxiliary Boiler to RCIC
12-741	Auxiliary Boiler to RCIC
43-28A	RHR Suppression Chamber Sample Lines
43-28B	RHR Suppression Chamber Sample Lines
43-29A	RHR Suppression Chamber Sample Lines
43-29B	RHR Suppression Chamber Sample Lines
71-14	RCIC Turbine Exhaust
71-32	RCIC Vacuum Pump Discharge
71-580	RCIC Turbine Exhaust
71-592	RCIC Vacuum Pump Discharge
73-23	HPCI Turbine Exhaust
73-24	HPCI Turbine Exhaust Drain
73-603	HPCI Turbine Exhaust
73-609	HPCI Exhaust Drain
74-722	RHR
75-57	Suppression Chamber Drain
75-58	Suppression Chamber Drain

Amendment No. ~~57~~, ~~78~~, 99,

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 128 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 99 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

DOCKET NOS. 50-259, 50-260 AND 50-296

1.0 INTRODUCTION

By letter dated October 1, 1985 (TVA BFNP TS-213), the Tennessee Valley Authority (the licensee or TVA) requested amendments to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3. The proposed amendments would change the Technical Specifications to correct inconsistencies and typographical errors, and would add new surveillance requirements.

2.0 EVALUATION

The amendments would modify the Technical Specifications (TS) as follows:

(A) The Table of Contents, for Units 1 and 2, would be updated to delete listings for specifications which were deleted in previous amendments. This would be an editorial change only, having no safety significance and is acceptable.

(B) A reference on page 9 to the surveillance requirement in the Limiting Safety System Setting specification for the Unit 1 Average Power Range Monitor (APRM) would be corrected. The present incorrect reference to Section 4.1.B on page 32 (which specifies surveillance requirements for the Reactor Protection System Power Monitoring System), would be replaced by a reference to Section 4.5.L. which specifies surveillance requirements for the APRM scram set points on page 160A and is the correct reference. This would be an editorial change, having no safety significance, and is acceptable.

(C) Annual channel functional test requirements for the triaxial peak accelographs would be added to the seismic instrumentation surveillance requirements table for all units Limiting conditions for operation are specified for these instruments but no surveillance requirements are presently specified. This change is consistent with 10 CFR 50.36(c)(3) which requires that Technical Specifications include surveillance requirements as necessary to assure limiting conditions for operation will be met and is therefore acceptable.

(D) Grammatical and typographical errors in the surveillance requirements for the Unit 3 control rod system would be corrected. The errors originated in Amendment 56. The proposed changes would revise the wording to be consistent with Units 1 and 2. These changes are editorial, have no effect on safety, and are acceptable.

(E) Terminology used in the surveillance requirements for the Units 1, 2 and 3 Automatic Depressurization System would be revised to be consistent with the associated limiting condition for operation. In the surveillance requirement "When ... more than two" would be changed to "When ... three of the six." This is an editorial change having no effect on safety and is therefore acceptable.

(F) Section 4.6.B.6 of the TS for Units 1 and 2 (page 179) lists the surveillance requirements on the frequency for analyzing primary coolant for iodine -131 (I-131) to comply with the limiting conditions of operation (LCO) on I-131 in Section 3.6.B.6. For Unit 3, these same surveillance requirements are in Section 4.6.B.3 and the LCO on I-131 is in Section 3.6.B.5. Section 4.6.B.6d in the Units 1 and 2 TSs (Section 4.6.B.3.d in the Unit 3 TSs) requires additional analyses whenever the equilibrium iodine limits are exceeded and erroneously references limits in Section 3.6.B.4. The latter lists water chemistry limits on conductivity, chloride and pH but nothing on iodine. The licensee has proposed to correct this error by changing the references to the LCO section that lists iodine limits. There are no changes in any limits or frequencies of analyses. We have reviewed the changes and determined that they are necessary to correct an error and are acceptable.

(G) A typographical error would be corrected in the coolant chemistry limiting conditions for operation for Units 1 and 2. In Section 3.6.B.6, "steam lime" would be changed to "steam line." This change is typographical, has no safety significance, and is therefore acceptable.

(H) The referenced list of safety related snubbers for Units 1, 2 and 3, would be changed from "Surveillance Instruction BF SI 4.6.H" to Surveillance Instructions BF SI 4.6.H.-1 and -2." This change would reflect changes to plant procedures. Such changes may be made by the licensee in accordance with Technical Specification 6.3.B and are acceptable.

(I) Technical Specification 3.7.A.1, the limiting condition for operation for the pressure suppression chamber water level and temperature, for Units 1, 2 and 3, would be changed to delete a reference to exceptions in Section 3.7.A.2. There are no exceptions to the water level and temperature limitations specified in 3.7.A.2. This is an editorial change, has no safety significance, and is therefore acceptable.

(J) Unit 2 Table 3.7.A "Primary Containment Isolation Valves" would be revised to indicate that air compressor suction valve FCV-64-139 and air compressor discharge valve FCV-64-140 are normally closed and stay closed on an initiating signal. These valves open only when the air compressor is running and are thus best described as normally closed. A footnote describing operation of these valves would also be deleted. This change is descriptive only. It would make the Unit 2 Technical Specifications consistent with Units 1 and 3 and is acceptable.

(K) Unit 2 Table 3.7.D "Air Tested Isolation Valves" would be revised to describe valves 90-254B and 90-255 as radiation monitor suction valves. These valves are currently listed in Table 3.7.D as radiation monitor discharge valves. This change is descriptive only, would make the valve descriptions consistent with the valve nomenclature in the radiation monitoring system and is acceptable. The revised Unit 2 Technical Specifications would be consistent with the Technical Specifications for Units 1 and 3.

(L) Unit 3 Table 3.7.A (page 264A) would be revised to describe RCIC steam line drain valves FSV-71-6A and FSV-71-6B as normally open and going closed on an initiation signal. The function of the RCIC steam line drain system requires that these valves be normally open. This change would make the Table 3.7.A valve descriptions consistent with the associated isolation instrumentation and is acceptable. The revised Unit 3 Technical Specifications would be consistent with Units 1 and 2 Technical Specifications (page 252).

(M) For Units 1, 2 and 3 Table 3.7.E would be revised to describe valves 75-57 and 75-58, presently described as "Core Spray to auxiliary boiler" as "Suppression chamber drain valves." This change would make the Table 3.7.E descriptions consistent with Table 3.7.A and reflect actual plant nomenclature. These are editorial changes having no safety significance and are acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

The amendments involve changes in requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and provide additional plant surveillance. 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 the 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.

4.0 CONCLUSION

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: W. Long and R. Clark

Dated: March 31, 1986