FEB 24 1981

Distribution
Docket
ORB #2
Local PDR
NRC PDR
DEisenhut
RPurple
TNovak
RTedesco
GLainas
SNorris
RClark
TIppolito
OELD
IE (5)
TBarnhardt (12)
BScharf (10)
JWetmore
ACRS (16)
OPA (CMiles)
RDiggs
HDenton
JHeltemes, AEOD
NSIC
TERA

Docket Nos. 50-259
~~50-260~~
and 50-296

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

Dear Mr. Parris:

The Commission has issued the enclosed Amendment Nos. 67, 63 and 39 to Facility License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2, and 3. These amendments are in response to your letter of September 24, 1980 (TVA BFNP TS 151).

The amendments change the Technical Specifications to modify the bases for scram insertion times by specifying 290 milliseconds as the time period to be used in the analytical treatment of transients for the start of control rod motion.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 67 to DPR-33
2. Amendment No. 63 to DPR-52
3. Amendment No. 39 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures: See next page

no legal objections
make changes on
P1 & P3 of SER
Working Copy

8103130028

OFFICE	ORB #2	ORB #2	AD:OR	OELD	ORB #2		
SURNAME	SNorris	RClark:mjf	TNovak	JLaverty	TIppolito		
DATE	2/12/81	2/10/81	2/12/81	2/23/81	2/12/81		

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

February 24, 1981

Docket No. 50-259
50-260
and 50-296

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

Dear Mr. Parris:

The Commission has issued the enclosed Amendment Nos. 67, 63 and 39 to Facility License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2, and 3. These amendments are in response to your letter of September 24, 1980 (TVA BFNP TS 151).

The amendments change the Technical Specifications to modify the bases for scram insertion times by specifying 290 milliseconds as the time period to be used in the analytical treatment of transients for the start of control rod motion.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 67 to DPR-33
2. Amendment No. 63 to DPR-52
3. Amendment No. 39 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures: See next page

cc:

H. S. Sanger, Jr., Esquire
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E 11B 33C
Knoxville, Tennessee 37902

Mr. Ron Rogers
Tennessee Valley Authority
400 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Mr. Charles R. Christopher
Chairman, Limestone County Commission
P. O. Box 188
Athens, Alabama 35611

Ira L. Myers, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36104

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

Athens Public Library
South and Forrest
Athens, Alabama 35611

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

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street
Atlanta, Georgia 30308

Mr. Robert F. Sullivan
U. S. Nuclear Regulatory Commission
P. O. Box 1863
Decatur, Alabama 35602

Mr. John F. Cox
Tennessee Valley Authority
W9-D 207C
400 Commerce Avenue
Knoxville, Tennessee 37902

Mr. Herbert Abercrombie
Tennessee Valley Authority
P. O. Box 2000
Decatur, Alabama 35602

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNTI NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 67
License No. DPR-33

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

   A. The application for amendments by Tennessee Valley Authority (the licensee), dated September 24, 1980, 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 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. 67, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachments:
Changes to the Technical
Specifications

Date of Issuance: February 24, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 67

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Review Appendix as follows:

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

3/4
79/80
133/134
322/323
326/327

2. The underlined pages are those being changed; marginal lines on these pages indicate the area being revised. Overleaf pages are provided for convenience.

1.0 DEFINITIONS (cont'd)

E. Operable - Operability - A system, subsystem, train, component, or device shall be Operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

F. Operating - Operating means that a system or component is performing its intended functions in its required manner.

G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

H. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.

I. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212°F, system pressure less than 1055 psig, the main steam isolation valves closed and the mode switch in the Startup/Hot Standby position.

J. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

K. Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

L. Cold Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature equal to or less than 212°F.

M. Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided:

1. Startup/Hot Standby Mode - In this mode the reactor protection scram trips initiated by condenser low vacuum and main steam line isolation valve colsure, are bypassed when reactor pressure is less than 1055 psig, the reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15% high flux trip, and control rod withdrawal interlocks in service. This is often referred to as just Startup Mode. This is intended to imply the startup/Hot Standby position of the mode switch.

1.0 DEFINITIONS (Cont'd)

2. Run Mode - In this mode the reactor system pressure is at or above 825 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

3. Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 sec), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig.

4. Refuel Mode - With the mode switch in the refuel position interlocks are established so that one control rod only may be withdrawn when the Source Range Monitor indicate at least 3 cps and the refueling crane is not over the reactor; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

N. Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, corresponds to 3,440 MWt.

O. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant systems or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

P. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 37 H_2M - 76 - 39	Drywell H_2 Concentration	0.1 - 20%	(1)
1	H_2M - 76 - 38	Suppression Chamber H_2 Concentration	0.1 - 20%	(1)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential pressure	Indicator 0 to 2 psid	(1) (2) (3)

Amendment No. ~~38~~, 46

DEC 8 197

NOTES FOR TABLE 3.2.F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

(3) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Cold Condition within 24 hours.

3.3/4.3 BASES :

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10% of the control rods at 16-week intervals is adequate for determining the operability of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Browns Ferry Nuclear Plant.

The occurence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

In the analytical treatment of the transients which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of control rod motion.

This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from sensor and circuit delays after which the pilot scram solenoid deenergizes to 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds, rather than 120 milliseconds, are conservatively assumed for this time interval in the transient analyses and are also included in the allowable scram insertion times of Specification 3.3.C.

* In order to perform scram testing as required by specification 4.3.C.1, the relaxation of certain restraints in the rod sequence control system is required. Individual rod bypass switches may be used as described in specification 4.3.C.1.

The position of any rod bypassed must be known to be in accordance with rod withdrawal sequence. Bypassing of rods in the manner described in specification 4.3.C.1 will allow the subsequent withdrawal of any rod scrammed in the 100 percent to 50 percent rod densitry groups; however, it will maintain group notch control over all rods in the 50 percent density to preset power level range. In addition, RSCS will prevent movement of rods in the 50 percent density to preset power level range until the secammed rod has been withdrawn.

3.3/4.4 BASES:

D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% ΔK Deviations in core reactivity greater than 1% Δk are not expected and require thorough evaluation. One percent reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

References

1. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

Amendment No. ~~35~~, ~~47~~, 59

FEB 25 1980

3.11 FIRE PROTECTION SYSTEMS

E. If it becomes necessary to breach a fire stop, an attendant shall be posted on each aide of the open penetration until work is completed and the penetration is resealed.

F. The minimum in-plant fire protection organization and duties shall be as depicted in Figure 6.3-1.

4.11 FIRE PROTECTION SYSTEMS

3.11 FIRE PROTECTION SYSTEMS

G. A minimum of fifteen air masks and thirty 500 cubic inch air cylinders shall be available at all times except that a time period of 48 hours following emergency use is allowed to permit recharging or replacing.

H. A continuous fire watch shall be stationed in the immediate vicinity where work involving open flame welding, or burning is in progress.

I. There shall be no use of open flame, welding, or burning in the cable spreading room unless the reactor is in the cold shutdown condition.

4.11 FIRE PROTECTION SYSTEMS

3.11 BASES

The High Pressure Fire and CO_2 Fire Protection specifications are provided in order to meet the preestablished levels of operability during a fire in either or all of the three units. Requiring a patrolling fire watch with portable fire equipment if the automatic initiation is lost will provide (as does the automatic system) for early reporting and immediate fire fighting capability in the event of a fire occurrence.

The High Pressure Fire Protection System is supplied by four pumps (three electric driven and one diesel driven) aligned to the high pressure fire header. The reactors may remain in operation for a period not to exceed 7 days if three pumps are out of service. If at least two pumps are not made operable in seven days or if all pumps are lost during this seven day period, the reactors will be placed in the cold shutdown condition within 24 hours.

For the areas of applicability, the fire protection water distribution system minimum capacity of 2664 gpm at 250' head at the fire pump discharge consists of the following design loads:

1.	Sprinkler System (0.30 gpm/ft^2/4440 ft^2 area)	1332 gpm
2.	1 1/2" Hand Hose Lines	200 gpm
3.	Raw Service Water Load	1132 gpm
	TOTAL	2664 gpm

The CO_2 Fire Protection System is considered operable with a minimum of 8 1/2 tons (0.5 tank) CO_2 in storage for units 1 and 2; and a minimum of 3 tons (0.5 tank) CO_2 in storage for unit 3. An immediate and continuous fire watch in the cable spreading room or any diesel generator building area will be established if CO_2 fire protection is lost in this room and will continue until CO_2 fire protection is restored.

To assure close supervision of fire protection system activities, the removal from service of any component in either the High Pressure Fire System or the CO_2 Fire Protection System for any reason other than testing or emergency operations will require Plant Superintendent approval.

Early reporting and immediate fire fighting capability in the event of a fire occurrence will be provided (as with the automatic system) by requiring a patrolling fire watch if more than one detector for a given protected zone is inoperable.

A roving fire watch for areas in which automatic fire suppression systems are to be installed will provide additional interim fire protection for areas that have been determined to need additional protection.

The fire protectio… system is designed to supp… the required flow and pressure to an individual load listed on Table 3.11.A while maintaining a design raw service water load of 1132 gpm.

4.11 BASES

Periodic testing of both the High Pressure Fire System and the CO_2 Fire Protection System will provide positive indication of their operability. If only one of the pumps supplying the High Pressure Fire System is operable, the pump that is operable will be checked immediately and daily thereafter to demonstrate operability. If the CO_2 Fire Protection System becomes inoperable in the cable spreading room, one 125-pound (or larger) fire extinguishere will be placed at each entrance to the cable spreading room.

Annual testing of automatic valves and control devices is in accordance with NFPA code Vol. II, 1975, section 15, paragraph 6015. More frequent testing would require excessive automatic system inoperability, since there are a large number of automatic valves installed and various portions of the system must be isolated during an extended period of time during this test.

Wet fire header flushing, spray header inspection for blockage, and nozzle inspection for blockage will prevent, detect, and remove buildup of sludge or other material to ensure continued operability. System flushes in conjunction with the semiannual addition of biocide to the Raw Cooling Water System will help prevent the growth of crustaceans which could reduce nozzle discharge.

Semiannual tests of heat and smoke detectors are in accordance with the NFPA code.

With the exception of continuous strip heat detectors panels, all non-class A supervised detector circuits which provide alarm only are hardwired through conduits and/or cable trays from the detector to the main control room alarm panels with no active components between. Non-class A circuits also actuate the HPCI water-fog system, the CO_2 system in the diesel generator buildings, and isolate ventilation in shutdown board rooms. The test frequency and methods specified are justified for the following reasons:

1. An analysis was made of worst-case fire detection circuits at Browns Ferry to determine the probability of no undetected failure of the circuits occurring between system test times as specified in the surveillance requirements. A circuit is defined as the wire connections and components that affect transmission of an alarm signal between the fire detectors and the control room annunciator. Three circuits were analyzed which were representative of an alarm-only circuit, a water-fog circuit, and a CO_2 circuit. The spreading room B smoke detector was selected as the worst-case alarm-only circuit because it had the largest number of wires and connections in a single circuit. The HPCI water-fog circuit was selected for analysis because it is the only water-fog circuit in the area of applicability for technical specifications. The Standby Diesel Generator Room A CO_2

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 63
License No. DPR-52

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

   A. The application for amendments by Tennessee Valley Authority (the licensee) dated September 24, 1980, 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 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. 63, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to Technical
Specifications

Date of Issuance: February 24, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 63

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.

3/4
79/80
133/134
321/322
323/324
325/326

2. The underlined pages are those being changed; marginal lines on these pages indicate the area being revised. Overleaf pages are provided for convenience.

1.0 DEFINITIONS (cont'd)

E. Operable - Operability - A system, subsystem, train, component, or device shall be Operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

F. Operating - Operating means that a system or component is performing its intended functions in its required manner.

G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

H. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.

I. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212°F, system pressure less than 1055 psig, the main steam isolation valves closed and the mode switch in the Startup/Hot Standby position.

J. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

K. Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

L. Cold Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature equal to or less than 212°F.

M. Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided:

1. Startup/Hot Standby Mode - In this mode the reactor protection scram trips initiated by condenser low vacuum and main steam line isolation valve colsure, are bypassed when reactor pressure is less than 1055 psig, the reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15% high flux trip, and control rod withdrawal interlocks in service. This is often referred to as just Startup Mode. This is intended to imply the startup/Hot Standby position of the mode switch.

1.0 DEFINITIONS (Cont'd)

2. Run Mode - In this mode the reactor system pressure is at or above 825 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

3. Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 sec), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig.

4. Refuel Mode - With the mode switch in the refuel position interlocks are established so that one control rod only may be withdrawn when the Source Range Monitor indicate at least 3 cps and the refueling crane is not over the reactor; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

N. Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, corresponds to 3,440 MWt.

O. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant systems or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

P. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 94 H_2M - 76 - 104	Drywell and Torus Hydrogen Concentration	0.1 - 20%	(1)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential pressure	Indicator 0 to 2 psid	(1) (2) (5)

Amendment No. 36, 42, 58
NOV 12 1980

NOTES FOR TABLE 3.2.F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

(3) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Cold Condition within 24 hours.

3.3/4.3 BASES :

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10% of the control rods at 16-week intervals is adequate for determining the operability of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Browns Ferry Nuclear Plant.

The occurence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

In the analytical treatment of the transients which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of control rod motion.

This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from sensor and circuit delays after which the pilot scram solenoid deenergizes to 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds, rather than 120 milliseconds, are conservatively assumed for this time interval in the transient analyses and are also included in the allowable scram insertion times of Specification 3.3.C.

* In order to perform scram testing as required by specification 4.3.C.1, the relaxation of certain restraints in the rod sequence control system is required. Individual rod bypass switches may be used as described in specification 4.3.C.1.

The position of any rod bypassed must be known to be in accordance with rod withdrawal sequence. Bypassing of rods in the manner described in specification 4.3.C.1 will allow the subsequent withdrawal of any rod scrammed in the 100 percent to 50 percent rod densitry groups; however, it will maintain group notch control over all rods in the 50 percent density to preset power level range. In addition, RSCS will prevent movement of rods in the 50 percent density to preset power level range until the secammed rod has been withdrawn.

3.3/4.4 **BASES:**

D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% ΔK Deviations in core reactivity greater than 1%Δk are not expected and require thorough evaluation. One percent reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

References

1. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A, and Addenda.

3.11 FIRE PROTECTION SYSTEMS

D. A roving fire watch will tour each area in which automatic fire suppression systems are to be installed (as described in the "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2," Section X) at intervals no greater than 2 hours. A keyclock recording type system shall be used to monitor the routes of the roving fire watch. The patrol will be discontinued as the automatic suppression systems are installed and made operable for each specified area.

4.11 FIRE PROTECTION SYSTEMS

3. The class A supervised detector alarm circuits will be tested once each two months at the local panels.

4. The circuits between the local panels in 4.11.C.3 and the main control room will be tested monthly.

5. Smoke detector sensitivity will be checked in accordance with manufacturer's instruction annually.

D. A monthly walk-through by the Safety Engineer will be made to visually inspect the plant fire protection system for signs of damage, deterioration, or abnormal conditions which could jeopardize proper operation of the system.

3.11 FIRE PROTECTION SYSTEMS

E. If it becomes necessary to breach a fire stop, an attendant shall be posted on each aide of the open penetration until work is completed and the penetration is resealed.

F. The minimum in-plant fire protection organization and duties shall be as depicted in Figure 6.3-1.

4.11 FIRE PROTECTION SYSTEMS

3.11 FIRE PROTECTION SYSTEMS

G. A minimum of fifteen air masks and thirty 500 cubic inch air cylinders shall be available at all times except that a time period of 48 hours following emergency use is allowed to permit recharging or replacing.

H. A continuous fire watch shall be stationed in the immediate vicinity where work involving open flame welding, or burning is in progress.

I. There shall be no use of open flame, welding, or burning in the cable spreading room unless the reactor is in the cold shutdown condition.

4.11 FIRE PROTECTION SYSTEMS

TABLE 3.11.A

FIRE PROTECTION SYSTEM HYDRAULIC REQUIREMENTS

Station	Flow Required (gpm)	Residual Pressure (psig)
1. Reactor Building Roof		
A. Valve 26-849	200	65
B. Valve 26-889	200	65
2. Refuel Floor		
A. Valve 26-835	75	70
B. Valve 26-843	75	70
C. Valve 26-870	75	70
D. Valve 26-865	75	70
E. Valve 26-876	75	70
F. Valve 26-888	75	70
G. Valve 26-898	75	70
3. Cable Tray Fixed Spray		
A. Unit 1 - Station I	300	70
B. Unit 1 - Station II	200	70
C. Unit 1 - Station III	180	65
D. Unit 2 - Station II	200	70
E. Unit 2 - Station III	200	70
F. Unit 3 - Station II	200	70
G. Unit 3 - Station III	265	75
H. Turbine Building	30	55
4. Diesel Generator Buildings		
A. Valve 26-1032	75	70
B. Valve 26-1069	75	70
5. Pump Intake Station		
A. Valve 26-578	75	70

TABLE 3.11.A

FIRE PROTECTION SYSTEM HYDRAULIC REQUIREMENTS

Station	Flow Required (gpm)	Residual Pressure (psig)
6. Control Bay		
A. Valve 26-1076	75	70
7. Yard Loop (1)		
A. Hydrant at valve 0-26-526	500	65
B. Hydrant at valve 0-26-530	500	65
8. Cooling Tower Loop		
A. Hydrant at valve 0-26-1023-6	500	65

Note (1) Yard hydrants and the cooling tower hydrant are to be tested using the longest path for flow.

3.11 BASES

The High Pressure Fire and CO_2 Fire Protection specifications are provided in order to meet the preestablished levels of operability during a fire in either or all of the three units. Requiring a patrolling fire watch with portable fire equipment if the automatic initiation is lost will provide (as does the automatic system) for early reporting and immediate fire fighting capability in the event of a fire occurrence.

The High Pressure Fire Protection System is supplied by four pumps (three electric driven and one diesel driven) aligned to the high pressure fire header. The reactors may remain in operation for a period not to exceed 7 days if three pumps are out of service. If at least two pumps are not made operable in seven days or if all pumps are lost during this seven day period, the reactors will be placed in the cold shutdown condition within 24 hours.

For the areas of applicability, the fire protection water distribution system minimum capacity of 2664 gpm at 250' head at the fire pump discharge consists of the following design loads:

1.	Sprinkler System (0.30 gpm/ft^2/4440 ft^2 area)	1332 gpm
2.	1 1/2" Hand Hose Lines	200 gpm
3.	Raw Service Water Load	1132 gpm
	TOTAL	2664 gpm

The CO_2 Fire Protection System is considered operable with a minimum of 8 1/2 tons (0.5 tank) CO_2 in storage for units 1 and 2; and a minimum of 3 tons (0.5 tank) CO_2 in storage for unit 3. An immediate and continuous fire watch in the cable spreading room or any diesel generator building area will be established if CO_2 fire protection is lost in this room and will continue until CO_2 fire protection is restored.

To assure close supervision of fire protection system activities, the removal from service of any component in either the High Pressure Fire System or the CO_2 Fire Protection System for any reason other than testing or emergency operations will require Plant Superintendent approval.

Early reporting and immediate fire fighting capability in the event of a fire occurrence will be provided (as with the automatic system) by requiring a patrolling fire watch if more than one detector for a given protected zone is inoperable.

A roving fire watch for areas in which automatic fire suppression systems are to be installed will provide additional interim fire protection for areas that have been determined to need additional protection.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
License No. DPR-68

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

   A. The application for amendments by Tennessee Valley Authority (the licensee) dated September 24, 1980, 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 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. 39, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 24, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 39

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:

   3
   83
   136
   354
   355
   356

2. Marginal lines on the above pages indicate revised area.

I. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212°F, system pressure less than 1055 psig, the main steam isolation valves closed and the mode switch in the Startup/Hot Standby position.

J. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

K. Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

L. Cold Shutdown - The reactor is in the shutdown mode, the reactor coolant temperature equal to or less than 212°F, and the reactor vessel is vented to atmosphere.

M. Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided:

1. Startup/Hot Standby Mode - In this mode the reactor protection scram trips initiated by condenser low vacuum and main steam line isolation valve closure, are bypassed when reactor pressure is less than 1055 psig, the reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15% high flux trip, and control rod withdrawal interlocks in service. This is often referred to as just Startup Mode. This is intended to imply the Startup/Hot Standby position of the mode switch.

2. Run Mode - In this mode the reactor system pressure is at or above 825 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

3. Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 sec), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig.

NOTES FOR TABLE 3.2.F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

(3) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

(5) If the requirements of notes (1) and (2) cannot be met, and if one of the indications cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Cold Condition within 24 hours.

In the analytical treatment of the transients which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of control rod motion.

This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from the sensor and circuit delays after which the pilot scram solenoid deenergizes and 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds, rather than 120 milliseconds, are conservatively assumed for this time interval in the transient analyses and are also included in the allowable scram insertion times of Specification 3.3.C.

In order to perform scram time testing as required by specification 4.3.C.1, the relaxation of certain restraints in the rod sequence control system is required. Individual rod bypass switches may be used as described in specification 4.3.C.1.

The position of any rod bypassed must be known to be in accordance with rod withdrawal sequence. Bypassing of rods in the manner described in specification 4.3.C.1 will allow the subsequent withdrawal of any rod scrammed in the 100 percent to 50 percent rod density groups; however, it will maintain group notch control over all rods in the 50 percent to 0 percent rod density groups. In addition, RSCS will prevent movement of rods in the 50 percent density to a preset power level range until the scrammed rod has been withdrawn.

D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% ΔK. Deviations in core reactivity greater than 1% ΔK are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

References

1. General Electric Supplemental Reload Licensing Submittal for BFNP unit 3 Reload 2, NEDO-24199, July 1979.

3.11 FIRE PROTECTION SYSTEMS

E. If it becomes necessary to breach a fire stop, an attendant shall be posted on each aide of the open penetration until work is completed and the penetration is resealed.

F. The minimum in-plant fire protection organization and duties shall be as depicted in Figure 6.3-1.

4.11 FIRE PROTECTION SYSTEMS

3.11 FIRE PROTECTION SYSTEMS

G. A minimum of fifteen air masks and thirty 500 cubic inch air cylinders shall be available at all times except that a time period of 48 hours following emergency use is allowed to permit recharging or replacing.

H. A continuous fire watch shall be stationed in the immediate vicinity where work involving open flame welding, or burning is in progress.

I. There shall be no use of open flame, welding, or burning in the cable spreading room unless the reactor is in the cold shutdown condition.

4.11 FIRE PROTECTION SYSTEMS

3.11 BASES

The High Pressure Fire and CO_2 Fire Protection specifications are provided in order to meet the preestablished levels of operability during a fire in either or all of the three units. Requiring a patrolling fire watch with portable fire equipment if the automatic initiation is lost will provide (as does the automatic system) for early reporting and immediate fire fighting capability in the event of a fire occurrence.

The High Pressure Fire Protection System is supplied by four pumps (three electric driven and one diesel driven) aligned to the high pressure fire header. The reactors may remain in operation for a period not to exceed 7 days if three pumps are out of service. If at least two pumps are not made operable in seven days or if all pumps are lost during this seven day period, the reactors will be placed in the cold shutdown condition within 24 hours.

For the areas of applicability, the fire protection water distribution system minimum capacity of 2664 gpm at 250' head at the fire pump discharge consists of the following design loads:

1.	Sprinkler System (0.30 gpm/ft^2/4440 ft^2 area)	1332 gpm
2.	1 1/2" Hand Hose Lines	200 gpm
3.	Raw Service Water Load	1132 gpm
	TOTAL	2664 gpm

The CO_2 Fire Protection System is considered operable with a minimum of 8 1/2 tons (0.5 tank) CO_2 in storage for units 1 and 2; and a minimum of 3 tons (0.5 tank) CO_2 in storage for unit 3. An immediate and continuous fire watch in the cable spreading room or any diesel generator building area will be established if CO_2 fire protection is lost in this room and will continue until CO_2 fire protection is restored.

To assure close supervision of fire protection system activities, the removal from service of any component in either the High Pressure Fire System or the CO_2 Fire Protection System for any reason other than testing or emergency operations will require Plant Superintendent approval.

Early reporting and immediate fire fighting capability in the event of a fire occurrence will be provided (as with the automatic system) by requiring a patrolling fire watch if more than one detector for a given protected zone is inoperable.

A roving fire watch for areas in which automatic fire suppression systems are to be installed will provide additional interim fire protection for areas that have been determined to need additional protection.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NOS. 1, 2 AND 3

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

1.0 Introduction

By letter dated September 24, 1980 (TVA BFNP TS 151), 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, Unit Nos. 1, 2 and 3. The proposed amendments would change the Technical Specifications to modify the bases for scram insertion times by specifying 290 milliseconds as the time period to be used in the analytical treatment of transients for the start of control rod motion.

2.0 Discussion

The present Technical Specifications for Browns Ferry Units 1, 2 and 3 (BF-1, BF-2 and BF-3) in the bases for Section 3.3.C state that "in the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion". In the recent core reload anlayses performed by the General Electric Company (GE) for TVA, the transient analyses for all three Browns Ferry units have used a value of 290 milliseconds for transients which are assumed to scram on high neutron flux. The proposed change in the Technical Specifications is to change the 390 millisecond value in the present bases to 290 milliseconds to bring the bases into conformance with the core reload analyses currently being performed by G.E. Three other minor administrative changes to the Technical Specifications are included in these amendments; these are discussed in the following evaluation.

3.0 Evaluation

As noted above, the proposed change is to use 290 msec rather than 390 msec in certain transient analyses. The 290 msec has been accepted by NRC when we approved G.E.'s generic reload topical report. The 290 msec is adequate and conservative when compared to the typical time delay of about 210 msec estimated from scram test results. Approximately the first 90 msec of each of these time intervals results from the sensor and circuit delays after which the pilot scram solenoid deenergizes and 120 msec later the control rod motion is estimated to actually begin. However, to be conservative, 200 msec rather than 120 msec is assumed for this time interval in the transient analyses and this is also the value used to develop the 290 msec as the allowable scram insertion times of Specification 3.3.C. The staff concludes that the proposed change is both appropriate and necessary to bring the bases of the BF-1, BF-2 and BF-3 Technical Specifications into line with the present supplemental core reload analyses.

Several administrative type changes to the Technical Specifications are also included herein to correct errors or changes in plant conditions. On page 4 of the definitions for each unit, the pressure at which the mode switch can be in the run mode is changed from 850 psig to 825 psig to be consistent with the bases in section 2.1.G and H (p 24 for BF-1 and 2, p 23 for BF-3). As stated in the bases, "the low pressure isolation of the main steam lines at 825 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the ORM and APRM high neutron flux scrams".

Another change included herein is to correct an erroneous reference in a note to Table 3.2.F. This table lists the "surveillance instrumentation" in containment, such as drywell temperature and pressure, suppression chamber water level and temperature, etc. Note 3 to this table states that "either the requirements of 3.5.H shall be complied with or .....". However, 3.5.H. (p 158 for BF-1 and 2, p 163 for BF-3) specifies that "whenever the core spray system's LPCI, HPCI or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled". This is obviously an erroneous reference. In correcting this note, rather than refer the reader to another section of the Technical Specifications, the specifically required action to be taken is spelled out - namely, that "if one of the indications cannot be restored in (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold condition within 24 hours".

The other two proposed changes relate to the fire protection system. As part of the fire restoration program, a diesel driven pump was added to the high pressure fire protection system. The present bases state that the system is supplied by three pumps. The proposed change is to reflect the condition that actually exists - that the system is supplied by four pumps (three electric driven and one diesel driven) aligned to the high pressure fire header. On February 13, 1980, we issued Amendment Nos. 58, 53 and 31 to Facility License Nos. DPR-33, DPR-52 and DPR-68 to increase the duties and functions of the Browns Ferry Nuclear Safety Review Board (BFNSRB). Section 6.2.A.8 lists the audits that are required to be performed under the cognizance of the NSRB. Sections 6.2.A.8.i and 6.2.A.8.j specify two of the required audits of the fire protection program. These two requirements are also included, verbatim, as a limiting condition of operation (LCO) in Section 3.11 of the Technical Specifications relating to the fire protection system. The appropriate location for specifying audits is in the Administrative Controls section of the Technical Specifications. Since the specific requirements are already incorporated in Section 6.2.8, the duplication in Section 3.11 can be properly eliminated.

4.0 Environmental Considerations

We have determined that the amendments do 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 amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 5.15(d)(4) that an environmental impact statement or negative declaration and enviornmental impact appraisal need not be prepared in connection with the issuance of the amendments.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: February 24, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY

OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 67 to Facility Operating License No. DPR-33, Amendment No. 63 to Facility Operating License No. DPR-52, and Amendment No. 39 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

These amendments change the Technical Specifications to modify the bases for scram insertion times by specifying 290 milliseconds as the time period to be used in the analytical treatment of transients for the start of control rod motion.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and 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) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated September 24, 1980, (2) Amendment No. 67 to License No. DPR-33, Amendment No. 63 to License No. DPR-52, and Amendment No. 39 to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) 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 24th day of February 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing