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Dockets Nos. 50-259/260 & 296

February 24, 1977

Tennessee Valley Authority
ATTN: Mr. Godwin Williams, Jr.
Manager of Power
818 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 29, 26 and 4 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments consist of changes to the Technical Specifications in response to your request of January 26, 1977 as supplemented February 7, 1977.

These amendments change the Technical Specifications to decrease the main steam line isolation pressure setpoint. You were previously notified of these changes by telephone and by telecopy on February 7, 1977.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 29 to DPR-33
2. Amendment No. 26 to DPR-52
3. Amendment No. 4 to DPR-68
4. Safety Evaluation
5. Federal Register Notice

cc w/enclosures:
See next page

ML020040243

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SURNAME x27433:tsh	~~TVWambach~~		~~ASchwencer~~	SBurwell		
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cc: H. S. Sanger, Jr. , Esquire
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State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36104

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Knoxville, Tennessee 37902

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
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 January 26, 1977, as supplemented February 7, 1977, 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. 29, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective February 7, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 24, 1977

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 29 TO FACILITY LICENSE NO. DPR-33

AMENDMENT NO. 26 TO FACILITY LICENSE NO. DPR-52

AMENDMENT NO. 4 TO FACILITY LICENSE NO. DPR-68

DOCKETS NOS. 50-259, 50-260 & 50-296

Revise Appendix A of Units 1 & 2 as follows:

> Remove pages 10, 24, 55 and 112 and replace with identically numbered pages.

Revise Appendix A of Unit 3 as follows:

> Remove pages 13, 23, 57 and 110 and replace with identically numbered pages.

2.1 FUEL CLADDING INTEGRITY

or core coolant flow is less than 10% of rated, the core thermal power shall not exceed 823 MWt (about 25% of rated thermal power).

C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 in. above the top of the normal active fuel zone.

2.1 FUEL CLADDING INTEGRITY

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

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)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of 2.63 the setting shall be modified as follows:

$$S_{RB} \leq \{0.66W + 42\%\}\frac{2.63}{MTPF}$$

where:

MTPF = The value of the existing maximum total peaking factor

C.	Scram and isolation—reactor low water level	≥ 538 in. above vessel zero
D.	Scram—turbine stop valve closure	≤ 10 percent valve closure
E.	Scram—turbine control valve	
	1. Fast closure	Upon trip of the fast acting solenoid valves
	2. Loss of control oil pressure	≥ 550 psig
F.	Scram—low condenser vacuum	≥ 23 inches Hg vacuum
G.	Scram—main steam line isolation	≤ 10 percent valve closure
H.	Main steam isolation valve closure—nuclear system low pressure	≥ 825 psig

2.1 BASES

2. Scram on loss of control oil pressure

The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the generator load rejection scram, since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection scram on fast closure of the turbine control valves, a scram has been added to the reactor protection system, which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in pressure or neutron flux occurs. The transient response is very similar to that resulting from the generator load rejection.

F. Main Condenser Low Vacuum Scram

To protect the main condenser against overpressure, a loss of condenser vacuum initiates automatic closure of the turbine stop valves and turbine bypass valves. To anticipate the transient and automatic scram resulting from the closure of the turbine stop valves, low condenser vacuum initiates a scram. The low vacuum scram set point is selected to initiate a scram before the closure of the turbine stop valves is in

G. & H. Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram

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. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. 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 IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel – Reactor Low Water Level (6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel – Reactor High Pressure	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel – Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 490" above vessel zero.	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel – High Drywell Pressure (6) (PS-64-56A-D)	≤ 2 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
2	Instrument Channel – High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated full power background	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel – Low Pressure Main Steam Line	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel – High Flow Main Steam Line	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation

3.2 BASES

LPCI loop selection logic and trips the recirculation pumps. The low reactor water level instrumentation that is set to trip when reactor water level is 17.7" (378" above vessel zero) above the top of the active fuel (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus the results given above are applicable here also.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting of 200°F for the main steam line tunnel detector is low enough to detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR. An alarm, with a nominal set point of 1.5 x normal full power background, is provided also.

Pressure instrumentation is provided to close the main steam isolation valves in Run Mode when the main steam line pressure drops below 825 psig.

1.1 FUEL CLADDING INTEGRITY

D. Shutdown Condition

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 in. above the top of the normal active fuel zone.

2.1 FUEL CLADDING INTEGRITY

C.	Scram and isolation reactor low water level	≥ 538 in. above vessel zero
D.	Scram--turbine stop valve closure	≤ 10 percent valve closure
E.	Scram--turbine control valve	
	1. Fast closure--Upon trip of the fast acting solenoid valves	
	2. Loss of control oil pressure	≥ 550 psig
F.	Scram--low condenser vacuum	≥ 23 inches Hg vacuum
G.	Scram--main steam line isolation	≤ 10 percent valve closure
H.	Main steam isolation valve closure --nuclear system low pressure	≤ 825 psig
I.	Core spray and LPCI actuation--reactor low water level	≥ 378 in. above vessel zero
J.	HPCI and RCIC actuation--reactor low water level	≥ 490 in. above vessel zero
K.	Main steam isolation valve closure--reactor low water level	≥ 490 in. above vessel zero

setting and the fact that control valve closure time is approximately twice as long as that for the stop valves means that resulting transients, while similar, are less severe than for stop-valve closure. No fuel damage occurs, and reactor system pressure does not exceed the relief valve set point, which is approximately 280 psi below the safety limit.

2. Scram on loss of control oil pressure

The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the generator load rejection scram, since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection scram on fast closure of the turbine control valves, a scram has been added to the reactor protection system, which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in pressure or neutron flux occurs. The transient response is very similar to that resulting from the generator load rejection.

F. Main Condenser Low Vacuum Scram

To protect the main concenser against overpressure, a loss of condenser vacuum initiates automatic closure of the turbine stop valves and turbine bypass valves. To anticipate the transient and automatic scram resulting from the closure of the turbine stop valves, low condenser vacuum initiates a scram. The low vaccum scram set point is selected to initiate a scram before the closure of the turbine stop valves is initiated.

G. & H. Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram

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. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the STARTUP

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level (6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 490" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)	≤ 2 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
2	Instrument Channel - High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated full power background	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Low Pressure Main Steam Line	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Main Steam Line Tunnel High Temperature	≤ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation.

Pressure instrumentation is provided to close the main steam isolation valves in Run Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic, and all sensors are required to be operable.

High temperature in the vicinity of the HPCI equipment is sensed by 4 sets of 4 bimetallic temperature switches. The 16 temperature switches are arranged in 2 trip systems with 8 temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" water for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.05. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control

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. 26
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 January 26, 1977, as supplemented February 7, 1977, 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. 26 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective February 7, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 24, 1977

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. 4
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 January 26, 1977, as supplemented February 7, 1977, 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. 4 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective February 7, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 24, 1977

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 26 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 4 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR GENERATING PLANT, UNITS 1, 2 & 3

Introduction

By letter dated January 26, 1977, as supplemented by letter dated February 7, 1977, the Tennessee Valley Authority (TVA) proposed changes to the Technical Specifications appended to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68. The proposed changes involve a reduction in the main steam line low pressure isolation setpoint and are the same as changes previously approved by the Nuclear Regulatory Commission (NRC) for other nuclear plants. (Hatch 1, Brunswick 2, Monticello and Oyster Creek).

Discussion and Evaluation

Main Steam Line Pressure Isolation Set Point Reduction

Installation of the main steam line low pressure sensors was required to provide reactor isolation in the event of an abnormal transient associated with the failure of the initial turbine pressure regulator in the open direction. This reactor isolation function was provided to limit the duration and severity of system depressurization so that no significant thermal stresses are imposed on the primary system. No credit was taken for these low pressure sensors in any of the other postulated abnormal operating transients or accidents. The current isolation set point is 850 psig; the proposed setpoint is 825 psig.

TVA referenced Edwin I. Hatch Nuclear Plant Unit 1 (50-321) submittal dated October 9, 1975 which provided a bounding analysis for a reduction in the main steam line low pressure setpoint from 880 psig to 825 psig. The NRC staff has reviewed the Hatch I analysis and has determined that it is applicable to TVA's proposed changes. In both cases (Hatch and Browns Ferry) the additional temperature decrease and subsequent reactor vessel thermal stresses, resulting from the additional pressure reduction during the abnormal transient, are negligible. Because reduction of the low pressure isolation setpoint would not have significant effects on

previously analyzed transients, we have concluded that the proposed change is acceptable.

Environmental Consideration

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

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the 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.

Date: February 24, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS 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 Amendments Nos. 29, 26and 4 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68, respectively, 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 (the facility) located in Limestone County, Alabama. The amendments are effective February 7, 1977.

These amendments change the Technical Specifications to decrease the main steam line isolation pressure setpoint.

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 January 26, 1977 as supplemented February 7, 1977, (2) Amendments Nos. 29, 26 and 4 to Licenses Nos. DPR-33, DPR-52 and DPR-68, respectively 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 Operating Reactors.

Dated at Bethesda, Maryland, this 24th day of February 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors