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

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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. 68, 64 and 40 to Facility Licenses 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 March 1, 1979, as supplemented by your letter of August 7, 1979 (TVA BFNP TS 122). These amendments change the Technical Specifications to clarify calibration requirements for the Local Power Range Monitors, reduce the pressure at which scram time surveillance testing may be conducted and remove a preoperational startup test requirement which is no longer applicable.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 68 to DPR-33
2. Amendment No. 64 to DPR-52
3. Amendment No. 40 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures:
See page 2

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DATE	2/ /81	2/13/81	2/ /81	/ /81	/ /81		

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

February 27, 1981

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. 68, 64 and 40 to Facility Licenses 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 March 1, 1979, as supplemented by your letter of August 7, 1979 (TVA BFNP TS 122). These amendments change the Technical Specifications to clarify calibration requirements for the Local Power Range Monitors, reduce the pressure at which scram time surveillance testing may be conducted and remove a preoperational startup test requirement which is no longer applicable.

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. 68 to DPR-33
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cc w/enclosures:
See page 2

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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. 64
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 March 1, 1979, as supplemented by letter dated August 7, 1979, 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. 64, 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 27, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 64

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:

39/40
41/42
47/48
123/124
239/240
241/242

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

NOTES FOR TABLE 4.1.A

1. Initially the minimum frequency for the indicated tests shall be once per month.

2. A description of the three groups is included in the Bases of this specification.

3. Functional tests are not required when the systems are not required to be operable or are operating (i.e., already tripped). If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the monthly functional test program.

6. The functional test of the flow bias network is performed in accordance with Table 4.2.C.

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns (6)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once every 7 days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/operating cycle
LPRM Signal	B	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Pressure Standard	Every 3 Months
High Water Level in Scram Discharge Volume	A	Note (5)	Note (5)
Turbine Condenser Low Vacuum	A	Standard Vacuum Source	Every 3 Months
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 Months
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Every 6 Months
Turbine Control Valve - Loss of Oil Pressure	A	Standard Pressure Source	Once/operating cycle
Turbine Stop Valve Closure	A	Note (5)	Note (5)

NOTES FOR TABLE 4.1.B

1. A description of three groups is included in the bases of this specification.

2. Calibrations are not required when the systems are not required to be operable or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an operable status.

3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made each refueling outage.

4. Maximum frequency required is once per week.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled shutdowns, overlap between the IRM's and APRM's will be verified.

7. The Flow Bias Signal Calibration will consist of calibrating the sensors, flow converters, and signal offset networks during each operating cycle. The instrumentation is an analog type with redundant flow signals that can be compared. The flow comparator trip and upscale will be functionally tested according to Table 4.2.C to ensure the proper operating during the operating cycle. Refer to 4.1 Bases for further explanation of calibration frequency.

8. A complete tip system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100% power.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE - 279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure, turbine stop valve closure and loss of condenser vacuum are discussed in Specification 2.1 and 2.2.

The frequency of calibration of the APRM Flow Biasing Network has been established as each refueling outage. There are several instruments which must be calibrated and it will take several hours to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRM's resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month a drift of .4% would occur and thus providing for adequate margin. For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.A and 4.1.B indicates that two instrument channels have not been included in the latter table. These are: mode switch in shutdown and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable, i.e., the switch is either on or off.

The ratio of Core Maximum Fraction of Limiting Power Density (MFLPD) to Fraction of Rated Power (FRP) shall be checked out once per day to determine if the APRM scram requires adjustment. This will normally be done by checking the APRM readings. Only a small number of control rods are moved daily during steady-state operation and thus the ratio is not expected to change significantly.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of CMFLPD, CPR, MAPLHGR and R ratio are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core Tip traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle prior to reaching 100 percent power.

3.3.B Control Rods

b. During the shutdown procedure no rod movement is permitted between the testing performed above 20% power and the reinstatement of the RSCS restraints at or above 20% power. Alignment of rod groups shall be accomplished prior to performing the tests.

c. Whenever the reactor is in the startup or run modes below 20% rated power the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

A second licensed operator may not be used in leiu of the RWM during scram time testing in the startup or run modes below 20 percent of rated thermal power.

d. If Specifications 3.3.B.3.a through .c cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 20% rated power, it shall be brought to a shutdown condition immediately.

4.3.B Control Rods

a. The capability of the RSCS to properly fulfill its function shall be verified by the following tests:

Sequence portion - Select a sequence and attempt to withdraw a rod in the remaining sequences. Move one rod in a sequence and select the remaining sequences and attempt to move a rod in each. Repeat for all sequences.

Group notch portion - For each of the six comparator circuits go through test initiate; comparator inhibit; verify; reset. On seventh attempt test is allowed to continue until completion is indicated by illumination of test complete light.

b. The capability of the Rod Worth Minimizer (RWM) shall be verified by the following checks:

1. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified before reactor startup or shutdown.

2. The RWM computer on line diagnostic test shall be successfully performed.

3. Prior to startup, proper annunciation of the selection error of at least one out-of-sequence control rod shall be verified.

4. Prior to startup, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

5. Prior to obtaining 20% rated power during rod insertion at shutdown, verify the latching of the proper rod group and proper annunciation after insert errors.

3.3.B Control Rods

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:

   a. Both RBM channels shall be operable: or

   b. Control rod withdrawal shall be blocked.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.500

4.3.B Control Rods

c. When required, the presence of a second licensed operator to verify the following of the correct rod program shall be verified.

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and at least once per 24 hours thereafter.

C. Scram Insertion Times

1. After each refueling outage all operable rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to exceeding 40% power. Below 20% power, only rods in those sequences (A_{12} and A_{34} or B_{12} and B_{34}) which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram time tested. The sequence restraints imposed upon the control rods in the 100-50 percent rod density groups to the preset power level may be removed by use of the individual bypass switches associated with those control rods which are fully or partially withdrawn and are not within the 100-50 percent rod density groups. In order to bypass a rod, the actual rod axial position must be known; and the rod must be in the correct in-sequence position.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

4. If these conditions cannot be met, the reactor shall be placed in a condition for which the standby gas treatment system is not required.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

c. When one train of the standby gas treatment system becomes inoperable the other two trains shall be demonstrated to be operable within 2 hours and daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.C **Secondary Containment**	4.7.C **Secondary Containment**
1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.	1. Secondary containment surveillance shall be performed as indicated below:

3.7.C Secondary Containment

2. If reactor zone secondary containment integrity cannot be maintained the following conditions shall be met:

   a. The reactor shall be made subcritical and Specification 3.3.A shall be met.

   b. The reactor shall be cooled down below 212°F and the reactor coolant system vented.

   c. Fuel movement shall not be permitted in the reactor zone.

   d. Primary containment integrity maintained.

3. Secondary containment integrity shall be maintained in the refueling zone, except as specified in 3.7.C.4.

4.7.C Secondary Containment

   a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (< 5 mph) conditions with a system inleakage rate of not more than 12,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

2. After a secondary containment violation is determined the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4-inch of water negative pressure under calm wind conditions.

7. Secondary Containment

4. If refueling zone secondary containment cannot be maintained the following conditions shall be met:

a. Handling of spent fuel and all operations over spent fuel pools and open reactor wells containing fuel shall be prohibited.

b. The standby gas treatment system suction to the refueling zone will be blocked except for a controlled leakage area sized to assure the achieving of a vacuum of at least 1/4-inch of water and not over 3 inches of water in all three reactor zones.

Primary Containment Isolation Valves

1. During reactor power operation, all isolation valves listed in Table 3.7.A and all reactor coolant system instrument line flow check valves shall be operable except as specified in 3.7.D.2.

4.7.C Secondary Containment

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

b. At least once per quarter:

(1) All normally open power operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 68 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 Introduction

By letter dated March 1, 1979 (TVA BFNP TS 122) and supplemented at our request by letter dated August 7, 1979, 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 Nos. 1, 2 and 3. The proposed amendments would revise the Technical Specifications appended to the above Facility Operating Licenses to clarify calibration requirements for the Local Power Range Monitors (LPRMs), reduce the pressure at which scram time surveillance testing may be conducted and remove a preoperational start-up test requirement which has long since been completed and is no longer appropriate.

2.0 Discussion

The proposed changes on pages 40, 41, 47 and 48 for Units 1 and 2 and on pages 39, 40, 46 and 47 for Unit 3 consist of adding an explanatory note to Table 4.1.B, changing LPRM to APRM in 4.1 Bases, and changing 4.1 Bases for clarification of LPRM-APRM requirements. The purpose of these changes is to correct previous typographical errors and to clarify LPRM calibration requirements.

The proposed change on page 124 of the Technical Specifications for Units 1 and 2 (page 128 for Unit 3) is to change the pressure at which the control rods may be tested. The present Technical Specifications require that "after each refueling outage, all operable (control) rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 950 psig. TVA proposes to change the test pressure to 800 psig. The proposed change would allow post refuel outage control rod drive scram timing to be conducted in parallel with the vessel hydrostatic leak test, thus saving about one day in the start-up test sequence.

The proposed changes on pages 240 and 241 of the Technical Specifications for Units 1 and 2 (pages 251 and 252 for Unit 3) is to delete section 4.7.C.1a from the Technical Specifications and to reletter the remaining paragraph from b to a. This specification is no longer applicable to Browns Ferry as all preoperational tests are completed and the requirement to test secondary containment integrity once per cycle is specified in 4.7.C-1b (to become 4.7.C-1a).

3.0 Evaluation

3.1 LPRM Calibration

Table 4.1.B (p. 40 for Units 1 and 2, p. 39 for Unit 3) lists the minimum calibration frequencies for the reactor protection system (SCRAM) instrument channels. This table specifies that the "LPRM Signal" is to be calibrated every 1000 effective full power hours using the TIP System traverse data. The reason for this requirement and the time interval is explained in the bases. "The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. This is compensated for in the APRM system by calibrating every 7 days using heat balance data and by calibrating individual LPRM's every 1000 effective full-power hours using TIP traverse data." To clarify how this calibration is accomplished, TVA proposes to add a new explanatory note (Note 8) to 4.1.B which states: "A complete TIP System traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100% power." TVA also proposes to significantly amplify the bases (p. 48) for the calibration requirement to explain in more detail the procedures to be used to periodically check the LPRM, APRM and process computer readings.

The proposed changes to the Technical Specifications do not change any present requirement nor the time intervals for performing the required calibration. The purpose of the proposed changes are to clarify how the calibration has been accomplished and how TVA proposes to continue to meet the calibration requirement.

Based on our review of the licensee's submittal, we conclude that the modifications to the Technical Specifications are correct, consistent with the requirements of IEEE Std. 279-1971, and therefore, are acceptable.

3.2 Scram Time Test Pressure

As noted above, TVA proposes to change Section 4.3.C.1 of the Technical Specifications to allow scram time surveillance testing to be performed at 800 psig rather than 950 psig as is now required. In support of this proposed change, TVA has presented data which show that scram time surveillance at the lower pressure is conservative, i.e., the scram time measured at 800 psig is greater than that at 950 psig. The licensee has presented data for the 20% and 90% control rod drive scram insertion points for four control rods from each of the three Browns Ferry Units. The 20% insertion point is the most significant to the limiting, pressurization transient. These insertion data also demonstrate that conservative scram insertion will be assured through the range of scram position points. Based on our evaluation of scram time uncertainties in Reference 1, we have concluded that these data show a systematic conservatism, i.e., the bias between the 800 and 950 psig measurements is substantially greater than scram time standard deviations. Thus, the measurements at 800 psig would be a conservative representation of scram insertion times and effectiveness under transient conditions and for all scram positions.

We and TVA have reviewed the conduct of the vessel hydrostatic leak test and have concluded that it would not affect scram time surveillance or vice-versa.

We conclude that the proposed change to the Technical Specifications is acceptable.

3.3 Preoperational Containment Leak Test

Section 4.7.C.1.a (P. 240 and 241) of the present Technical Specifications required that "a preoperational secondary containment capability [shall] be conducted .....". This test was completed prior to startup of each unit to insure that secondary containment met the design objectives. To insure that secondary containment integrity is maintained during operation, Section 4.7.C.1.b requires that "secondary containment capability to maintain 1/4 inch water vacuum under calm wind conditions with a system inleakage rate of not more than 12,000 cfm shall be demonstrated at each refueling outage prior to refueling". Inasmuch as preoperational requirements such as above are no longer apropos, our objective is to remove such requirements from the Technical Specifications. Accordingly, TVA has proposed to delete the obsolete Section 4.7.C.1.a and to renumber the current requirement from "b" to "a". We conclude that the proposed action is appropriate and in keeping with our objective.

4.0 Environmental Considerations

We have determined that these 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 these amendments involve an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 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 these 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 considerations (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 27, 1981

Reference

1. Letter from T. A. Ippolito (USNRC) to D. Arnold (Iowa Electric Light and Power Company), September 4, 1979, attached safety evaluation for Amendment No. 54.

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 LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 68 to Facility Operating License No. DPR-33, Amendment No. 64 to Facility Operating License No. DPR-52, and Amendment No. 40 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 clarify calibration requirements for the Local Power Range Monitor, reduce the pressure at which scram time surveillance testing may be conducted and remove a preoperational startup test requirement which is no longer applicable.

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 March 1, 1979, as supplemented by letter dated August 7, 1979, (2) Amendment No. 68 to License No. DPR-33, Amendment No. 64 to License No. DPR-52, and Amendment No. 40 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 27th day of February 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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