



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. 68
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 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-33 is hereby amended to read as follows:

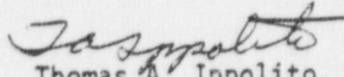
(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 68, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

108103130118

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. 68

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

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

2. The underlines 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 Basis 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 - Flow Bias Signal	B B	Heat Balance Calibrate Flow Bias Signal (7)	Once every 7 days 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)

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POOR ORIENTATION

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 scrambling 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 on 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

4.1 BASES

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, CPP, MAPLNCR 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 lieu 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.

L. LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

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:

<u>% Inserted From Fully Withdrawn</u>	<u>Avg. Scram Insertion Times (sec)</u>
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 (A₁₂ and A₃₄ or B₁₂ and B₃₄) 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

SURVEILLANCE REQUIREMENTS

3.7.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.

4.7.C Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

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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.



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

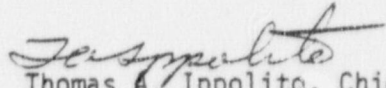
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendments by Tennessee Valley Authority (the licensee) dated March 1, 1979, as supplemented by letter dated August 7, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as 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)

or

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 scrambling 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 on 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

4.1 BASES

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, CPP, MAPLNCR 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.

3.2 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 lieu 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

2. 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.

3.3.8 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 REM 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:

<u>Z Inserted From Fully Withdrawn</u>	<u>Avg. Scram Insertion Times (sec)</u>
5	0.375
20	0.90
50	2.0
90	3.500

4.3.8 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 REM 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 (A₁₂ and A₃₄ or B₁₂ and B₃₄) 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.

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.

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

1. Secondary containment integrity shall be maintained in the reactor zone at all times except as specified in 3.7.C.2.

4.7.C Secondary Containment

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 leakage 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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 40
License No. DPR-68

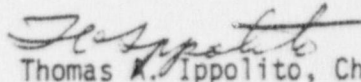
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendments by Tennessee Valley Authority (the licensee) dated March 1, 1979, as supplemented by letter dated August 7, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as 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-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. 40, 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. 40

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:

39
40
46
47
128
251
252

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

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 - Flow Bias Signal	B	Heat Balance	Once every 7 days
	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.

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 Table 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 (CMFLPD) 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, MAPLNGR 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 REACTIVITY CONTROL

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:

<u>% Inserted From Fully Withdrawn</u>	<u>Avg. Scram Insertion Times (sec)</u>
5	0.375
20	0.90
50	2.0
90	3.5

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

<u>% Inserted From Fully Withdrawn</u>	<u>Avg. Scram Insertion Times (sec)</u>
5	0.398
20	0.954
50	2.120
90	3.800

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

4.3 REACTIVITY CONTROL

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 (A₁₂ and A₃₄ or B₁₂ and B₃₄) 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.

2. At 16 week intervals, 10% of the operable control rod drives shall be scram timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

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.

3.7 CONTAINMENT SYSTEMS

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

3.7 CONTAINMENT SYSTEMS

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.

4.7 CONTAINMENT SYSTEMS

- 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.