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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
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 June 20, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 172, 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 its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jack n Donohew Jr for
Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 3, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 172

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1.0-7	1.0-7
1.0-8	1.0-8*
3.3/4.3-11	3.3/4.3-11
3.3/4.3-12	3.3/4.3-12*
3.10/4.10-3	3.10/4.10-3*
3.10/4.10-4	3.10/4.10-4
3.10/4.10-5	3.10/4.10-5
3.10/4.10-6	3.10/4.10-6
3.10/4.10-13	3.10/4.10-13
3.10/4.10-14	3.10/4.10-14*

1.0 DEFINITIONS (Cont'd)

Q. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

R. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

S. CORE ALTERATION - The addition, removal, relocation, or movement of fuel, sources, in-core instruments, or reactivity controls within the reactor pressure vessel with the head removed and fuel in the vessel. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a Core Alteration. Suspension of Core Alterations shall not preclude completion of the movement of a component to a safe conservative position.

T. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

U. Thermal Parameters

1. Minimum Critical Power Ratio (MCPR) - Minimum Critical Power Ratio (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

3. Core Maximum Fraction of Limiting Power Density (CMFLPD) - The highest ratio, for all fuel types in the core, of the maximum fuel rod power density (kW/ft) for a given fuel type to the limiting fuel rod power density (kW/ft) for that fuel type.

4. Average Planar Linear Heat Generation Rate (APLHGR) - The Average Planar Heat Generation Rate is applicable to a specific planar height and is equal to the sum of the linear heat generation rates for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

V. Instrumentation

1. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors.

2. Channel - A channel is an arrangement of the sensor(s) and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

3. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

4. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

5. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

6. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

7. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

8. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

9. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

LIMITING CONDITIONS FOR OPERATION

3.3.C. Scram Insertion Times

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:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
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.

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in the SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

SURVEILLANCE REQUIREMENTS

4.3.C. Scram Insertion Times

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.

D. Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.E. If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.	4.3.E. Surveillance requirements are as specified in 4.3.C and .D above.
F. Scram Discharge Volume (SDV)	F. Scram Discharge Volume (SDV)
1. The scram discharge volume drain and vent valves shall be operable any time that the reactor protection system is required to be operable except as specified in 3.3.F.2.	1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.
	1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.
2. In the event any SDV drain or vent valve becomes INOPERABLE, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is operable.	2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated operable immediately and weekly thereafter.
3. If redundant drain or vent valves become INOPERABLE, the reactor shall be in HOT STANDBY CONDITION within 24 hours.	3. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A. Refueling Interlocks	4.10.A. Refueling Interlocks
6. A maximum of two non-adjacent control rods may simultaneously be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance without removing the fuel from the cells provided the following conditions are satisfied:	6. Prior to performing control rod or control rod drive maintenance on two control cells simultaneously without removing the fuel from the cells, two SROs shall verify that the requirements of Specification 3.10.A.6 are satisfied.
a. The reactor mode switch shall be locked in the REFUEL position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be OPERABLE.	
b. All directional control valves for remaining control rods shall be disarmed electrically except as specified in 3.10.A.7 and sufficient margin to criticality shall be demonstrated.	
c. The two maintenance cells must be separated by more than two control cells in any direction.	
d. An appropriate number of SRMs are available as defined in Specification 3.10.B.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A. **Refueling Interlocks**	4.10.A. **Refueling Interlocks**
7. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied: a. The reactor mode switch is locked in the REFUEL position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be OPERABLE.	7. With the mode selection switch in the REFUEL or SHUTDOWN mode, no more than one control rod may be withdrawn without first removing fuel from the cell except as specified in 4.10.A.6. Any number of rods may be withdrawn once verified by two licensed operators that the fuel has been removed from each cell.
B. **Core Monitoring**	B. **Core Monitoring**
1. During core alterations, except as specified in 3.10.B.2, two SRMs (FLCs) shall be OPERABLE, one in and one adjacent to any quadrant where fuel or control rods are being moved. For an SRM (FLC) to be considered OPERABLE, the following shall be satisfied: a. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core	Prior to making any alterations to the core, the SRMs (FLCs) shall be functionally tested and checked for neutron response. Thereafter, while required to be OPERABLE, the SRMs will be checked daily for response.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.B. Core Monitoring 3.10.B.1.a. (Cont'd) alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.) b. When one or more fuel assemblies are in the core, except as specified in 3.10.B.2, the SRM (FLC) shall have a minimum indicated reading of 3 cps while monitoring the loaded assembly (assemblies) with all rods fully inserted in the core.	

3.10/4.10 CORE ALTERATIONS

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.B. Core Monitoring	4.10.B Core Monitoring
2. During a complete core removal, the SRMs shall have an initial minimum count rate of 3 cps prior to fuel removal. With all rods fully inserted and rendered electrically disarmed and inoperable, once the SRM count rate decreases below 3 cps, the SRMs will no longer be required to be OPERABLE during fuel removal. Individual control rods outside the periphery of the then existing fuel matrix may be electrically armed and moved for maintenance after all fuel in the cell containing (controlled by) that control rod have been removed from the reactor core.	

3.10 BASES (Cont'd)

B. Core Monitoring

The SRMs are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRMs (FLCs) one in and one adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of three counts per second provides assurance that neutron flux is being monitored and ensures that startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident. During a full core reload, the fuel will be loaded in control cells that are continuous to previously loaded control cells. This provided coupling of the loaded fuel matrix which is being monitored by the SRMs (FLCs).

Under the special condition of removing the full core with all control rods inserted and electrically disarmed, it is permissible to allow SRM count rate to decrease below three counts per second. All fuel moves during core unloading will reduce reactivity. It is expected that the SRMs will drop below three counts per second before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When sufficient fuel has been removed to the spent fuel storage pool to drop the SRM count rate below 3 cps, SRMs will no longer be required to be operable. Requiring the SRMs to be functionally tested prior to fuel removal assures that the SRMs will be operable at the start of fuel removal. The daily response check of the SRMs ensures their continued operability until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

REFERENCES

1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

2. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)

3.10 **BASES** (Cont'd)

C. **Spent Fuel Pool Water**

The design of the spent fuel storage pool provides a storage location for approximately 140 percent of the full core load of fuel assemblies in the reactor building which ensures adequate shielding, cooling, and reactivity control of irradiated fuel. An analysis has been performed which shows that a water level at or in excess of eight and one-half feet over the top of the stored assemblies will provide shielding such that the maximum calculated radiological doses do not exceed the limits of 10 CFR 20. The normal water level provides 14-1/2 feet of additional water shielding. The capacity of the skimmer surge tanks is available to maintain the water level at its normal height for three days in the absence of additional water input from the condensate storage tanks. All penetrations of the fuel pool have been installed at such a height that their presence does not provide a possible drainage route that could lower the normal water level more than one-half foot.

The fuel pool cooling system is designed to maintain the pool water temperature less than 125°F during normal heat loads. If the reactor core is completely unloaded when the pool contains two previous discharge batches, the temperature may increase to greater than 125°F. The RHR system supplemental fuel pool cooling mode will be used under these conditions to maintain the pool temperature to less than 125°F.

3.10.D/4.10.D BASES

Reactor Building Crane

The reactor building crane and 125-ton hoist are required to be operable for handling of the spent fuel in the reactor building. The controls for the 125-ton hoist are located in the crane cab. The five-ton has both cab and pendant controls.

A visual inspection of the load-bearing hoist wire rope assures detection of signs of distress or wear so that corrections can be promptly made if needed.

The testing of the various limits and interlocks assures their proper operation when the crane is used.

3.10.E/4.10.E

Spent Fuel Cask

The spent fuel cask design incorporates removable lifting trunnions. The visual inspection of the trunnions and fasteners prior to attachment to the cask assures that no visual damage has occurred during prior handling. The trunnions must be properly attached to the cask for lifting of the cask and the visual inspection assures correct installation.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 175
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 June 20, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 175, 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 its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jack N. Donohew Jr. for

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 3, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 175

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1.0-7	1.0-7
1.0-8	1.0-8*
3.3/4.3-11	3.3/4.3-11
3.3/4.3-12	3.3/4.3-12*
3.10/4.10-3	3.10/4.10-3*
3.10/4.10-4	3.10/4.10-4
3.10/4.10-5	3.10/4.10-5
3.10/4.10-6	3.10/4.10-6
3.10/4.10-13	3.10/4.10-13
3.10/4.10-14	3.10/4.10-14*

Q. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

R. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

S. CORE ALTERATION - The addition, removal, relocation, or movement of fuel, sources, incore instruments, or reactivity controls within the reactor pressure vessel with the head removed and fuel in the vessel. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a Core Alteration. Suspension of Core Alterations shall not preclude completion of the movement of a component to a safe conservative position.

T. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

U. Thermal Parameters

1. Minimum Critical Power Ratio (MCPR) - Minimum Critical Power Ratio (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

3. Core Maximum Fraction of Limiting Power Density (CMFLPD) - The highest ratio, for all fuel types in the core, of the maximum fuel rod power density (kW/ft) for a given fuel type to the limiting fuel rod power density (kW/ft) for that fuel type.

4. Average Planar Linear Heat Generation Rate (APLHGR) - The Average Planar Heat Generation Rate is applicable to a specific planar height and is equal to the sum of the linear heat generation rates for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

1.0 DEFINITIONS (Cont'd)

V. Instrumentation

1. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors.

2. Channel - A channel is an arrangement of the sensor(s) and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

3. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

4. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

5. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

6. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

7. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

8. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

9. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

LIMITING CONDITIONS FOR OPERATION

3.3.C. Scram Insertion Times

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:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
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.

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

SURVEILLANCE REQUIREMENTS

4.3.C. Scram Insertion Times

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.

D. Reactivity Anomalies

During the STARTUP test program and STARTUP following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.E. If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.	4.3.E. Surveillance requirements are as specified in 4.3.C and .D above.
F. Scram Discharge Volume (SDV)	F. Scram Discharge Volume (SDV)
1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.	1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.
	1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.
2. In the event any SDV drain or vent valve becomes INOPERABLE, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.	2. When it is determined that any SDV drain or vent valve is INOPERABLE, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. If redundant drain or vent valves become INOPERABLE, the reactor shall be in HOT STANDBY CONDITION within 24 hours.	3. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A. Refueling Interlocks	4.10.A. Refueling Interlocks
6. A maximum of two non-adjacent control rods may simultaneously be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance without removing the fuel from the cells provided the following conditions are satisfied:	6. Prior to performing control rod or control rod drive maintenance on two control cells simultaneously without removing the fuel from the cells, two SROs shall verify that the requirements of Specification 3.10.A.6 are satisfied.
a. The reactor mode switch shall be locked in the REFUEL position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be OPERABLE.	
b. All directional control valves for remaining control rods shall be disarmed electrically except as specified in 3.10.A.7 and sufficient margin to criticality shall be demonstrated.	
c. The two maintenance cells must be separated by more than two control cells in any direction.	
d. An appropriate number of SRMs are available as defined in Specification 3.10.B.	

LIMITING CONDITIONS FOR OPERATION

3.10.A. Refueling Interlocks

7. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied:

   a. The reactor mode switch is locked in the REFUEL position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be OPERABLE.

B. Core Monitoring

1. During core alterations, except as specified in 3.10.B.2, two SRMs (FLCs) shall be OPERABLE, one in and one adjacent to any quadrant where fuel or control rods are being moved. For an SRM (FLC) to be considered OPERABLE, the following shall be satisfied:

   a. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core

SURVEILLANCE REQUIREMENTS

4.10.A. Refueling Interlocks

7. With the mode selection switch in the REFUEL or SHUTDOWN mode, no more than one control rod may be withdrawn without first removing fuel from the cell except as specified in 4.10.A.6. Any number of rods may be withdrawn once verified by two licensed operators that the fuel has been removed from each cell.

B. Core Monitoring

Prior to making any alterations to the core, the SRMs (FLCs) shall be functionally tested and checked for neutron response. Thereafter, while required to be OPERABLE, the SRMs will be checked daily for response.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.B. Core Monitoring 3.10.B.1.a. (Cont'd) alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.) b. When one or more fuel assemblies are in the core, except as specified in 3.10.B.2, the SRM (FLC) shall have a minimum indicated reading of 3 cps while monitoring the loaded assembly (assemblies) with all rods fully inserted in the core.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.B. Core Monitoring	4.10.B Core Monitoring
2. During a complete core removal, the SRMs shall have an initial minimum count rate of 3 cps prior to fuel removal. With all rods fully inserted and rendered electrically disarmed and inoperable, once the SRM count rate decreases below 3 cps, the SRMs will no longer be required to be OPERABLE during fuel removal. Individual control rods outside the periphery of the then existing fuel matrix may be electrically armed and moved for maintenance after all fuel in the cell containing (controlled by) that control rod have been removed from the reactor core.	

3.10 **BASES** (Cont'd)

B. Core Monitoring

The SRMs are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRMs (FLCs) one in and one adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of three counts per second provides assurance that neutron flux is being monitored and ensures that startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident. During a full core reload, the fuel will be loaded in control cells that are contiguous to previously loaded control cells. This provides coupling of the loaded fuel matrix which is being monitored by the SRMs (FLCs).

Under the special condition of removing the full core with all control rods inserted and electrically disarmed, it is permissible to allow SRM count rate to decrease below three counts per second. All fuel moves during core unloading will reduce reactivity. It is expected that the SRMs will drop below three counts per second before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When sufficient fuel has been removed to the spent fuel storage pool to drop the SRM count rate below 3 cps, SRMs will no longer be required to be operable. Requiring the SRMs to be functionally tested prior to fuel removal assures that the SRMs will be operable at the start of fuel removal. The daily response check of the SRMs ensures their continued operability until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

REFERENCES

1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

2. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)

C. Spent Fuel Pool Water

The design of the spent fuel storage pool provides a storage location for approximately 140 percent of the full core load of fuel assemblies in the reactor building which ensures adequate shielding, cooling, and reactivity control of irradiated fuel. An analysis has been performed which shows that a water level at or in excess of eight and one-half feet over the top of the stored assemblies will provide shielding such that the maximum calculated radiological doses do not exceed the limits of 10 CFR 20. The normal water level provides 14-1/2 feet of additional water shielding. The capacity of the skimmer surge tanks is available to maintain the water level at its normal height for three days in the absence of additional water input from the condensate storage tanks. All penetrations of the fuel pool have been installed at such a height that their presence does not provide a possible drainage route that could lower the normal water level more than one-half foot.

The fuel pool cooling system is designed to maintain the pool water temperature less than 125°F during normal heat loads. If the reactor core is completely unloaded when the pool contains two previous discharge batches, the temperature may increase to greater than 125°F. The RHR system supplemental fuel pool cooling mode will be used under these conditions to maintain the pool temperature to less than 125°F.

D. Reactor Building Crane

The reactor building crane and 125-ton hoist are required to be operable for handling of the spent fuel in the reactor building. The controls for the 125-ton hoist are located in the crane cab. The five-ton has both cab and pendant controls.

A visual inspection of the load-bearing hoist wire rope assures detection of signs of distress or wear so that corrections can be promptly made if needed.

The testing of the various limits and interlocks assures their proper operation when the crane is used.

E. Spent Fuel Cask

The spent fuel cask design incorporates removable lifting trunnions. The visual inspection of the trunnions and fasteners prior to attachment to the cask assures that no visual damage has occurred during prior handling. The trunnions must be properly attached to the cask for lifting of the cask and the visual inspection assures correct installation.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 143
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 June 20, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 143, 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 its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jack N. Donohew Jr. for
Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 3, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 143

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1.0-7	1.0-7
1.0-8	1.0-8*
3.10/4.10-3	3.10/4.10-3*
3.10/4.10-4	3.10/4.10-4
3.10/4.10-5	3.10/4.10-5
3.10/4.10-6	3.10/4.10-6
3.10/4.10-11	3.10/4.10-11*
3.10/4.10-12	3.10/4.10-12

1.0 DEFINITIONS (Cont'd)

Q. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

R. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

S. CORE ALTERATION - The addition, removal, relocation, or movement of fuel, sources, incore instruments, or reactivity controls within the reactor pressure vessel with the head removed and fuel in the vessel. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a Core Alteration. Suspension of Core Alterations shall not preclude completion of the movement of a component to a safe conservative position.

T. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

U. Thermal Parameters

1. Minimum Critical Power Ratio (MCPR) - Minimum Critical Power Ratio (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

3. Core Maximum Fraction of Limiting Power Density (CMFLPD) - The highest ratio, for all fuel types in the core, of the maximum fuel rod power density (kW/ft) for a given fuel type to the limiting fuel rod power density (kW/ft) for that fuel type.

4. Average Planar Linear Heat Generation Rate (APLHGR) - The Average Planar Heat Generation Rate is applicable to a specific planar height and is equal to the sum of the linear heat generation rates for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

1.0 **DEFINITIONS** (Cont'd)

V. Instrumentation

1. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors.

2. Channel - A channel is an arrangement of the sensor(s) and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

3. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

4. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

5. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

6. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

7. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

8. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

9. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A. Refueling Interlocks	4.10.A. Refueling Interlocks
6. A maximum of two non-adjacent control rods may be simultaneously withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance without removing the fuel from the cells provided the following conditions are satisfied:	6. Prior to performing control rod or control rod drive maintenance on two control cells simultaneously without removing the fuel from the cells, two SROs shall verify that the requirements of Specification 3.10.A.6 are satisfied.
a. The reactor mode switch shall be locked in the REFUEL position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be OPERABLE.	
b. All directional control valves for remaining control rods shall be disarmed electrically except as specified in 3.10.A.7 and sufficient margin to criticality shall be demonstrated.	
c. The two maintenance cells must be separated by more than two control cells in any direction.	
d. An appropriate number of SRMs are available as defined in Specification 3.10.B.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A. **Refueling Interlocks**	4.10.A. **Refueling Interlocks**
7. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied: a. The reactor mode switch is locked in the REFUEL position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be OPERABLE.	7. With the mode selector switch in the REFUEL or SHUTDOWN mode, no more than one control rod may be withdrawn without first removing fuel from the cell except as specified in 4.10.A.6. Any number of rods may be withdrawn once verified by two licensed operators that the fuel has been removed from each cell.
B. **Core Monitoring**	B. **Core Monitoring**
1. During core alterations, except as specified in 3.10.B.2, two SRMs (FLCs) shall be OPERABLE, one in and one adjacent to any quadrant where fuel or control rods are being moved. For an SRM (FLC) to be considered OPERABLE, the following shall be satisfied: a. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core	Prior to making any alterations to the core, the SRMs (FLCs) shall be functionally tested and checked for neutron response. Thereafter, while required to be OPERABLE, the SRMs will be checked daily for response.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.B. Core Monitoring 3.10.B.1.a. (Cont'd) alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.) b. When one or more fuel assemblies are in the core, except as specified in 3.10.B.2, the SRM (FLC) shall have a minimum indicated reading of 3 cps while monitoring the loaded assembly (assemblies) with all rods fully inserted in the core.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.B. Core Monitoring	4.10.B Core Monitoring
2. During a complete core removal, the SRMs shall have an initial minimum count rate of 3 cps prior to fuel removal. With all rods fully inserted and rendered electrically disarmed and inoperable, once the SRM count rate decreases below 3 cps, the SRMs will no longer be required to be OPERABLE during fuel removal. Individual control rods outside the periphery of the then existing fuel matrix may be electrically armed and moved for maintenance after all fuel in the cell containing (controlled by) that control rod have been removed from the reactor core.	

and the refueling platform provide redundant methods of preventing inadvertent criticality even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 1,500 lbs, in comparison to the load-trip setting of 1,000 lbs. Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 400-lb load-trip setting on these hoists is adequate to trip the interlock when one of the more than 600-lb fuel bundles is being handled.

During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time without removing fuel from the cells. The maintenance is performed with the mode switch in the refuel position to provide the refueling interlocks normally available during refueling operations. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated and that all remaining control rods have their directional control valves electrically disarmed ensures that inadvertent criticality cannot occur during this maintenance. The adequacy of the shutdown margin is verified by demonstrating that at least 0.38 percent Δk shutdown margin is available. Disarming the directional control valves does not inhibit control rod scram capability.

Specification 3.10.A.7 allows unloading of a significant portion of the reactor core. This operation is performed with the mode switch in the REFUEL position to provide the refueling interlocks normally available during refueling operations. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod provides primary reactivity control for the fuel assemblies in the cell associated with that control rod.

Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core. The requirements for SRM operability during these core alterations assure sufficient core monitoring.

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

3.10 **BASES** (Cont'd)

B. Core Monitoring

The SRMs are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two OPERABLE SRMs (FLCs) one in and one adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of three counts per second provides assurance that neutron flux is being monitored and insures that startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident. During a full core reload, the fuel will be loaded in control cells that are contiguous to previously loaded control cells. This provides coupling of the loaded fuel matrix which is being monitored by the SRMs (FLCs).

Under the special condition of removing the full core with all control rods inserted and electrically disarmed, it is permissible to allow SRM count rate to decrease below three counts per second. All fuel moves during core unloading will reduce reactivity. It is expected that the SRMs will drop below three counts per second before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When sufficient fuel has been removed to the spent fuel storage pool to drop the SRM count rate below 3 cps, SRMs will no longer be required to be operable. Requiring the SRMs to be functionally tested prior to fuel removal assures that the SRMs will be OPERABLE at the start of fuel removal. The daily response check of the SRMs ensures their continued operability until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

References

1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

2. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)