UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 9, 1993

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

Dr. Mark O. Medford, Vice President
Nuclear Assurance, Licensing & Fuels
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Dr. Medford:

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS REGARDING REFUELING INTERLOCKS AND CORE MONITORING (TS 324) (TAC NOS. M84699, M84700, AND M84701)

The Commission has issued the enclosed Amendment Nos. 194 , 209 , and 166 to Facility Operating Licenses Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2, and 3, respectively. These amendments are in response to your application dated October 9, 1992, as supplemented March 31, 1993, requesting changes in surveillance requirements associated with certain refueling equipment, and changes to limiting conditions for operation and surveillance requirements associated with core reactivity monitoring during refueling operations.

The staff notes that TVA has verbally committed to revise the Technical Specification Bases to incorporate a discussion of "fueled region," consistent with the information provided in the supplemental information letter dated March 31, 1993. The staff requests that TVA make this revision before the next refueling of any Browns Ferry unit.

CP-1

DF01

A copy of the NRC's Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Frederick J. Hebdon, Director for
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.194 to License No. DPR-33
2. Amendment No.209 to License No. DPR-52
3. Amendment No.166 to License No. DPR-68
3. Safety Evaluation

cc w/enclosures:
See next page

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w/ sentence added to p. 1 of SE and other noted changes.

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ATTN: Dr. Mark O. Medford

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 194
License No. DPR-33

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 9, 1992, as supplemented on March 31, 1993, 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. 194 , 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 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

~~Frederick~~ J. Hebdon, Director for
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: **April 9, 1993**

ATTACHMENT TO LICENSE AMENDMENT NO. 194

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. Overleaf* and spillover** pages are provided to maintain document completeness.

REMOVE	INSERT
iii	iii
iv	iv*
1.0-7	1.0-7
1.0-8	1.0-8*
3.10/4.10-1	3.10/4.10-1
3.10/4.10-2	3.10/4.10-2
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**
3.10/4.10-13	3.10/4.10-13
3.10/4.10-14	3.10/4.10-14*
3.10/4.10-15	3.10/4.10-15
	3.10/4.10-16
6.0-3	6.0-3
6.0-4	6.0-4*

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 - CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, intermediate range monitors, traversing in-core probes, or special movable detectors (including undervessel replacement) is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe location.

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

3.10 CORE ALTERATIONS

Applicability

Applies to the fuel handling and core reactivity limitations.

Objective

To ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

Specification

A. Refueling Interlocks

1. The reactor mode switch shall be locked in the REFUEL position during CORE ALTERATIONS. The required refueling equipment interlocks shall be OPERABLE during in-vessel fuel movement with equipment associated with the interlocks except as specified in 3.10.A.6 and 3.10.A.7 below.

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

Applicability

Applies to the periodic testing of those refueling equipment interlocks and instrumentation required during refueling and CORE ALTERATIONS.

Objective

To verify the OPERABILITY of instrumentation and refueling equipment interlocks required during refueling and CORE ALTERATIONS.

Specification

A. Refueling Interlocks

1. Prior to any fuel handling with the head off the vessel, the following required refueling equipment interlocks shall be functionally tested:

   a. All rods inserted

   b. Refueling platform positioned near or over the core

   c. Refueling platform main hoist is fuel loaded

   d. Fuel grapple is not full up

   e. One rod withdrawn

3.10.A. Refueling Interlocks

2. Fuel shall not be loaded into the reactor core unless all control rods are fully inserted.

3. The fuel grapple hoist load switch shall be set at $\leq$ 1,000 lbs.

4. If the frame-mounted auxiliary hoist, the monorail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the load limit switch on the hoist to be used shall be set at $<$ 400 lbs.

4.10.A. Refueling Interlocks

4.10.A.1 (Continued)

* f. Refueling platform frame-mounted hoist is fuel loaded

* g. Monorail hoist is fuel loaded

* h. Service platform hoist is fuel loaded

They shall be tested at weekly intervals thereafter until no longer required. They shall also be tested following any repair work associated with the interlocks.

*NOTE: These interlocks are required to be OPERABLE only when the associated equipment is used for in-vessel fuel movement.

2. No additional surveillance required.

3. No additional surveillance required.

4. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

3.10.A. Refueling Interlocks

5. Maintenance may be performed on a single control rod or control rod drive without removing the fuel in the control cell if the following conditions are met:

   a. The requirements of Specification 3.10.A.1 are met, and

   b. All control rods diagonally and face adjacent to the maintenance rod are fully inserted and have their directional control valves electrically disarmed.

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:

   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 required refueling equipment interlocks shall be OPERABLE.

SURVEILLANCE REQUIREMENTS

4.10.A. Refueling Interlocks

5. Prior to performing control rod or control rod drive maintenance on a control cell without removing fuel assemblies the surveillance requirements of Specification 4.10.A.1 shall be performed and all rods face adjacent and diagonally adjacent to the maintenance rod shall be electrically disarmed per Specification 3.10.A.5.b.

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.

LIMITING CONDITIONS FOR OPERATION

3.10.A. Refueling Interlocks

3.10.A.6 (Cont'd)

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.

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 required refueling equipment interlocks shall be OPERABLE.

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.

LIMITING CONDITIONS FOR OPERATION

3.10.B. Core Monitoring

1. During CORE ALTERATIONS, except as specified in 3.10.B.2, two SRMs (FLCs) shall be OPERABLE. 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 ALTERATIONS in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)

   b. Verify an OPERABLE SRM (FLC) is located in:

      1. The fueled region;

      2. The quadrant where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region; and

      3. A core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region.

   Note: One SRM (FLC) may be used to satisfy more than one of the above.

SURVEILLANCE REQUIREMENTS

4.10.B. Core Monitoring

1. Prior to making any CORE ALTERATIONS, the SRMs (FLCs) shall be functionally tested and checked for neutron response.

2. Note: Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM (FLC) and no other fuel assemblies in the associated core quadrant.

   Once per 12 hours during CORE ALTERATIONS, verify that the associated SRM (FLC) is reading $\geq$ 3 cps with a signal-to-noise ratio $\geq$ 3:1.

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

A. Refueling Interlocks

The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations. The interlocks prevent an inadvertent criticality during refueling operations when the reactivity potential of the core is being altered.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

The refueling interlocks reinforce operational procedures that prohibit taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment.

The refueling interlocks include circuitry which senses the condition of the refueling equipment and the control rods. Depending on the sensed condition, interlocks are actuated which prevent the movement of the refueling equipment or withdrawal of control rods (rod block).

Circuitry is provided which senses the following conditions.

1. All rods inserted
2. Refueling platform positioned near or over the core
3. Refueling platform main hoist is fuel loaded
4. Fuel grapple not full up
5. One rod withdrawn
6. * Refueling platform frame-mounted hoist is fuel loaded
7. * Refueling platform monorail hoist is fuel loaded
8. * Service platform hoist is fuel loaded

When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. When the mode switch is in the refuel position only one control rod can be withdrawn. The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor is

* The refueling platform frame-mounted, monorail and the service platform fuel-loaded hoist interlocks are required to be OPERABLE only when utilized for in-vessel fuel movements.

subcritical even when the highest worth control rod is fully withdrawn. The combination of refueling interlocks for control rods 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.

3.10 BASES (Cont'd)

3.10.A (Cont'd)

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

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. Each SRM (FLC) is not required to read $\geq$ 3 cps until after four fuel assemblies have been loaded adjacent to the SRM (FLC) if no other fuel assemblies are in the associated core quadrant. These four locations are adjacent to the SRM dry tube. When utilizing FLCs, the FLCs will be located such that the required count rate is achieved without exceeding the SRM upscale setpoint. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical.

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 once per 12 hours verification of the SRM count rate and signal-to-noise ratio 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.

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.

3.10.F Spent Fuel Cask Handling - Refueling Floor

Although single failure protection has been provided in the design of the 125-ton hoist drum shaft, wire ropes, hook and lower block assembly on the reactor building crane, the limiting of lift height of a spent fuel cask controls the amount of energy available in a dropped cask accident when the cask is over the refueling floor.

An analysis has been made which shows that the floor and support members in the area of cask entry into the decontamination facility can satisfactorily sustain a dropped cask from a height of three feet.

The yoke safety links provide single failure protection for the hook and lower block assembly and limit cask rotation. Cask rotation is necessary for decontamination and the safety links are removed during decontamination.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all required refueling equipment interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform, and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its function.

B. Core Monitoring

Requiring the SRMs to be functionally tested prior to any CORE ALTERATION assures that the SRMs will be OPERABLE at the start of that alteration. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY.

REFERENCES

1. Fuel Pool Cooling and Cleanup System (BFNP FSAR Subsection 10.5)

2. Spent Fuel Storage (BFNP FSAR Subsection 10.3)

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6.2.2 (Cont.)

d. Two licensed reactor operators shall be in the control room during any cold startups, while shutting down the reactor, and during recovery from unit trip. In addition, a person holding a senior operator license shall be in the control room for that unit whenever it is in an operational mode other than cold shutdown or refueling.

e. A Health Physics Technician* shall be present at the facility at all times when there is fuel in the reactor.

f. Either a licensed SRO or licensed SRO limited to fuel handling who has no concurrent responsibilities during this operation shall be present during fuel handling and shall directly supervise all CORE ALTERATIONS.

g. Deleted.

*The Health Physics Technician may be absent from the facility for a period of time not to exceed 2 hours provided immediate action is taken to fill the required position.

Table 6.2.A
Minimum Shift Crew Requirements[b]

Position	Units in Operation 0	1	2[d]	3	Type of License
Senior Operator[a]	1	1	1	1	SRO
Senior Operator	0	1	2	2	SRO
Licensed Operators	3	3	3	3	RO or SRO
Additional Licensed Operators[c]	0	1	2	2	RO or SRO
Assistant Unit Operators (AUO)	4	4	5	5	None
Shift Technical Advisor (STA)	0	1	1	1	None
Health Physics Technician	1	1	1	1	None

Note for Table 6.2.A

a. A senior operator will be assigned responsibility for overall plant operation at all times there is fuel in any unit.

b. Except for the senior operator discussed in note "a", the shift crew composition may be one less than the minimum requirements of Table 6.2.A for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.A. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

c. One of the Additional Licensed Operators must be assigned to each control room with an operating unit.

d. The number of required licensed personnel, when the operating units are controlled from a common control room, are two senior operators and four operators.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 209
License No. DPR-52

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 9, 1992, as supplemented on March 31, 1993, 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. 209, 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 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 9, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 209

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. Overleaf* and spillover** pages are provided to maintain document completeness.

REMOVE	INSERT
iii	iii*
iv	iv
1.0-7	1.0-7
1.0-8	1.0-8*
3.10/4.10-1	3.10/4.10-1
3.10/4.10-2	3.10/4.10-2
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**
3.10/4.10-13	3.10/4.10-13
3.10/4.10-14	3.10/4.10-14*
3.10/4.10-15	3.10/4.10-15
	3.10/4.10-16
6.0-3	6.0-3
6.0-4	6.0-4*

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 - CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, intermediate range monitors, traversing in-core probes, or special movable detectors (including undervessel replacement) is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe location.

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

3.10 CORE ALTERATIONS

Applicability

Applies to the fuel handling and core reactivity limitations.

Objective

To ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

Specification

A. Refueling Interlocks

1. The reactor mode switch shall be locked in the REFUEL position during CORE ALTERATIONS. The required refueling equipment interlocks shall be OPERABLE during in-vessel fuel movement with equipment associated with the interlocks except as specified in 3.10.A.6 and 3.10.A.7 below.

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

Applicability

Applies to the periodic testing of those refueling equipment interlocks and instrumentation required during refueling and CORE ALTERATIONS.

Objective

To verify the OPERABILITY of instrumentation and refueling equipment interlocks required during refueling and CORE ALTERATIONS.

Specification

A. Refueling Interlocks

1. Prior to any fuel handling with the head off the vessel, the following required refueling equipment interlocks shall be functionally tested:

   a. All rods inserted

   b. Refueling platform positioned near or over the core

   c. Refueling platform main hoist is fuel loaded

   d. Fuel grapple is not full up

   e. One rod withdrawn

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A. Refueling Interlocks	4.10.A. Refueling Interlocks
	4.10.A.1 (Continued) * f. Refueling platform frame-mounted hoist is fuel loaded * g. Monorail hoist is fuel loaded * h. Service platform hoist is fuel loaded They shall be tested at weekly intervals thereafter until no longer required. They shall also be tested following any repair work associated with the interlocks. *NOTE: These interlocks are required to be OPERABLE only when the associated equipment is used for in-vessel fuel movement.
2. Fuel shall not be loaded into the reactor core unless all control rods are fully inserted.	2. No additional surveillance required.
3. The fuel grapple hoist load switch shall be set at ≤ 1,000 lbs.	3. No additional surveillance required.
4. If the frame-mounted auxiliary hoist, the monorail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the load limit switch on the hoist to be used shall be set at < 400 lbs.	4. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

3.10.A. Refueling Interlocks

5. Maintenance may be performed on a single control rod or control rod drive without removing the fuel in the control cell if the following conditions are met:

   a. The requirements of Specification 3.10.A.1 are met, and

   b. All control rods diagonally and face adjacent to the maintenance rod are fully inserted and have their directional control valves electrically disarmed.

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:

   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 required refueling equipment interlocks shall be OPERABLE.

SURVEILLANCE REQUIREMENTS

4.10.A. Refueling Interlocks

5. Prior to performing control rod or control rod drive maintenance on a control cell without removing fuel assemblies the surveillance requirements of Specification 4.10.A.1 shall be performed and all rods face adjacent and diagonally adjacent to the maintenance rod shall be electrically disarmed per Specification 3.10.A.5.b.

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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A. Refueling Interlocks	4.10.A. Refueling Interlocks

3.10.A. Refueling Interlocks

3.10.A.6 (Cont'd)

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.

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 required refueling equipment interlocks shall be OPERABLE.

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.

LIMITING CONDITIONS FOR OPERATION

3.10.B. Core Monitoring

1. During CORE ALTERATIONS, except as specified in 3.10.B.2, two SRMs (FLCs) shall be OPERABLE. 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 ALTERATIONS in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)

   b. Verify an OPERABLE SRM (FLC) is located in:

      1. The fueled region;

      2. The quadrant where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region; and

      3. A core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region.

   Note: One SRM (FLC) may be used to satisfy more than one of the above.

SURVEILLANCE REQUIREMENTS

4.10.B. Core Monitoring

1. Prior to making any CORE ALTERATIONS, the SRMs (FLCs) shall be functionally tested and checked for neutron response.

2. Note: Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM (FLC) and no other fuel assemblies in the associated core quadrant.

   Once per 12 hours during CORE ALTERATIONS, verify that the associated SRM (FLC) is reading $\geq$ 3 cps with a signal-to-noise ratio $\geq$ 3:1.

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

A. Refueling Interlocks

The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations. The interlocks prevent an inadvertent criticality during refueling operations when the reactivity potential of the core is being altered.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

The refueling interlocks reinforce operational procedures that prohibit taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment.

The refueling interlocks include circuitry which senses the condition of the refueling equipment and the control rods. Depending on the sensed condition, interlocks are actuated which prevent the movement of the refueling equipment or withdrawal of control rods (rod block).

Circuitry is provided which senses the following conditions.

1. All rods inserted
2. Refueling platform positioned near or over the core
3. Refueling platform main hoist is fuel loaded
4. Fuel grapple not full up
5. One rod withdrawn
6. * Refueling platform frame-mounted hoist is fuel loaded
7. * Refueling platform monorail hoist is fuel loaded
8. * Service platform hoist is fuel loaded

When the mode switch is in the REFUEL position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. When the mode switch is in the refuel position only one control rod can be withdrawn. The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor is

* The refueling platform frame-mounted, monorail and the service platform fuel-loaded hoist interlocks are required to be OPERABLE only when utilized for in-vessel fuel movements.

3.10 BASES (Cont'd)

subcritical even when the highest worth control rod is fully withdrawn. The combination of refueling interlocks for control rods 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.

3.10 <u>BASES</u> (Cont'd)

3.10.A (Cont'd)

<u>REFERENCES</u>

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

B. <u>Core Monitoring</u>

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. Each SRM (FLC) is not required to read $\geq$ 3 cps until after four fuel assemblies have been loaded adjacent to the SRM (FLC) if no other fuel assemblies are in the associated core quadrant. These four locations are adjacent to the SRM dry tube. When utilizing FLCs, the FLCs will be located such that the required count rate is achieved without exceeding the SRM upscale setpoint. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical.

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 once per 12 hours verification of the SRM count rate and signal-to-noise ratio 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.

<u>REFERENCES</u>

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.

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.

3.10.F Spent Fuel Cask Handling - Refueling Floor

Although single failure protection has been provided in the design of the 125-ton hoist drum shaft, wire ropes, hook and lower block assembly on the reactor building crane, the limiting of lift height of a spent fuel cask controls the amount of energy available in a dropped cask accident when the cask is over the refueling floor.

An analysis has been made which shows that the floor and support members in the area of cask entry into the decontamination facility can satisfactorily sustain a dropped cask from a height of three feet.

The yoke safety links provide single failure protection for the hook and lower block assembly and limit cask rotation. Cask rotation is necessary for decontamination and the safety links are removed during decontamination.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all required refueling equipment interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform, and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its function.

B. Core Monitoring

Requiring the SRMs to be functionally tested prior to any CORE ALTERATION assures that the SRMs will be OPERABLE at the start of that alteration. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY.

REFERENCES

1. Fuel Pool Cooling and Cleanup System (BFNP FSAR Subsection 10.5)

2. Spent Fuel Storage (BFNP FSAR Subsection 10.3)

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6.2.2 (Cont.)

d. Two licensed reactor operators shall be in the control room during any cold startups, while shutting down the reactor, and during recovery from unit trip. In addition, a person holding a senior operator license shall be in the control room for that unit whenever it is in an operational mode other than cold shutdown or refueling.

e. A Health Physics Technician* shall be present at the facility at all times when there is fuel in the reactor.

f. Either a licensed SRO or licensed SRO limited to fuel handling who has no concurrent responsibilities during this operation shall be present during fuel handling and shall directly supervise all CORE ALTERATIONS.

g. Deleted.

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*The Health Physics Technician may be absent from the facility for a period of time not to exceed 2 hours provided immediate action is taken to fill the required position.

Table 6.2.A
Minimum Shift Crew Requirements[b]

Position	Units in Operation				Type of License
	0	1	2[d]	3	
Senior Operator[a]	1	1	1	1	SRO
Senior Operator	0	1	2	2	SRO
Licensed Operators	3	3	3	3	RO or SRO
Additional Licensed Operators[c]	0	1	2	2	RO or SRO
Assistant Unit Operators (AUO)	4	4	5	5	None
Shift Technical Advisor (STA)	0	1	1	1	None
Health Physics Technician	1	1	1	1	None

Note for Table 6.2.A

a. A senior operator will be assigned responsibility for overall plant operation at all times there is fuel in any unit.

b. Except for the senior operator discussed in note "a", the shift crew composition may be one less than the minimum requirements of Table 6.2.A for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.A. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

c. One of the Additional Licensed Operators must be assigned to each control room with an operating unit.

d. The number of required licensed personnel, when the operating units are controlled from a common control room, are two senior operators and four operators.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 166
License No. DPR-68

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 9, 1992, and supplemented on March 31, 1993, 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 166, 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 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: **April 9, 1993**

ATTACHMENT TO LICENSE AMENDMENT NO. 166

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. Overleaf* and spillover** pages are provided to maintain document completeness.

REMOVE	INSERT
iii	iii
iv	iv*
1.0-7	1.0-7
1.0-8	1.0-8*
3.10/4.10-1	3.10/4.10-1
3.10/4.10-2	3.10/4.10-2
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*
3.10/4.10-9	3.10/4.10-9*
3.10/4.10-10	3.10/4.10-10
3.10/4.10-11	3.10/4.10-11**
3.10/4.10-12	3.10/4.10-12
3.10/4.10-13	3.10/4.10-13*
3.10/4.10-14	3.10/4.10-14
6.0-3	6.0-3
6.0-4	6.0-4*

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 - CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, intermediate range monitors, traversing in-core probes, or special movable detectors (including undervessel replacement) is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe location.

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

3.10 CORE ALTERATIONS

Applicability

Applies to the fuel handling and core reactivity limitations.

Objective

To ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

Specification

A. Refueling Interlocks

1. The reactor mode switch shall be locked in the REFUEL position during CORE ALTERATIONS. The required refueling equipment interlocks shall be OPERABLE during in-vessel fuel movement with equipment associated with the interlocks except as specified in 3.10.A.6 and 3.10.A.7 below.

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

Applicability

Applies to the periodic testing of those refueling equipment interlocks and instrumentation required during refueling and CORE ALTERATIONS.

Objective

To verify the OPERABILITY of instrumentation and refueling equipment interlocks required during refueling and CORE ALTERATIONS.

Specification

A. Refueling Interlocks

1. Prior to any fuel handling with the head off the vessel, the following required refueling equipment interlocks shall be functionally tested:

   a. All rods inserted

   b. Refueling platform positioned near or over the core

   c. Refueling platform main hoist is fuel loaded

   d. Fuel grapple is not full up

   e. One rod withdrawn

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A. Refueling Interlocks	4.10.A. Refueling Interlocks

4.10.A.1 (Continued)

* f. Refueling platform frame-mounted hoist is fuel loaded

* g. Monorail hoist is fuel loaded

* h. Service platform hoist is fuel loaded

They shall be tested at weekly intervals thereafter until no longer required. They shall also be tested following any repair work associated with the interlocks.

*NOTE: These interlocks are required to be OPERABLE only when the associated equipment is used for in-vessel fuel movement.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2. Fuel shall not be loaded into the reactor core unless all control rods are fully inserted.	2. No additional surveillance required.
3. The fuel grapple hoist load switch shall be set at ≤ 1,000 lbs.	3. No additional surveillance required.
4. If the frame-mounted auxiliary hoist, the monorail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the load limit switch on the hoist to be used shall be set at < 400 lbs.	4. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

3.10.A. Refueling Interlocks

5. Maintenance may be performed on a single control rod or control rod drive without removing the fuel in the control cell if the following conditions are met:

   a. The requirements of Specification 3.10.A.1 are met, and

   b. All control rods diagonally and face adjacent to the maintenance rod are fully inserted and have had their directional control valves electrically disarmed.

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:

   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 required refueling equipment interlocks shall be OPERABLE.

SURVEILLANCE REQUIREMENTS

4.10.A. Refueling Interlocks

5. Prior to performing control rod or control rod drive maintenance on a control cell without removing fuel assemblies the surveillance requirements of Specification 4.10.A.1 shall be performed and all rods face adjacent and diagonally adjacent to the maintenance rod shall be electrically disarmed per Specification 3.10.A.5.b.

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.

LIMITING CONDITIONS FOR OPERATION

3.10.A. Refueling Interlocks

3.10.A.6 (Cont'd)

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.

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 required refueling equipment interlocks shall be OPERABLE.

SURVEILLANCE REQUIREMENTS

4.10.A. Refueling Interlocks

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.

LIMITING CONDITIONS FOR OPERATION

3.10.B. Core Monitoring

1. During CORE ALTERATIONS, except as specified in 3.10.B.2, two SRMs (FLCs) shall be OPERABLE. 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 ALTERATIONS in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)

   b. Verify an OPERABLE SRM (FLC) is located in:

      1. The fueled region;

      2. The quadrant where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region; and

      3. A core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region.

   Note: One SRM (FLC) may be used to satisfy more than one of the above.

SURVEILLANCE REQUIREMENTS

4.10.B. Core Monitoring

1. Prior to making any CORE ALTERATIONS, the SRMs (FLCs) shall be functionally tested and checked for neutron response.

2. Note: Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM (FLC) and no other fuel assemblies in the associated core quadrant.

   Once per 12 hours during CORE ALTERATIONS, verify that the associated SRM (FLC) is reading $\geq$ 3 cps with a signal-to-noise ratio $\geq$ 3:1.

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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.D. Reactor Building Crane	4.10.D. Reactor Building Crane
E. Spent Fuel Cask	E. Spent Fuel Cask
1. Upon receipt, an empty fuel cask shall not be lifted until a visual inspection is made of the cask-lifting trunnions and fastening connection has been conducted.	1. Prior to attachment and lifting of an empty spent fuel cask from the shipping trailer, a visual inspection shall be conducted on the lifting trunnions and the fasteners used to connect the trunnion to the cask.
	2. A visual inspection shall be made of the assembled trunnion on the empty cask to insure proper assembly.
3.10.F Spent Fuel Cask Handling - Refueling Floor	
1. Administrative control shall be exercised to limit the height the spent fuel cask is raised above the refueling floor by the reactor building crane to 6 inches, except for entry into the cask decontamination chamber where height above the floor will be approximately 3 feet.	
2. The spent fuel cask yoke safety links shall be properly positioned at all times except when the cask is in the decontamination chamber.	

3.10 BASES

A. Refueling Interlocks

The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations. The interlocks prevent an inadvertent criticality during refueling operations when the reactivity potential of the core is being altered.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

The refueling interlocks reinforce operational procedures that prohibit taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment.

The refueling interlocks include circuitry which senses the condition of the refueling equipment and the control rods. Depending on the sensed condition, interlocks are actuated which prevent the movement of the refueling equipment or withdrawal of control rods (rod block).

Circuitry is provided which senses the following conditions.

1. All rods inserted
2. Refueling platform positioned near or over the core
3. Refueling platform main hoist is fuel loaded
4. Fuel grapple not full up
5. One rod withdrawn
6. * Refueling platform frame-mounted hoist is fuel loaded
7. * Refueling platform monorail hoist is fuel loaded
8. * Service platform hoist is fuel loaded

When the mode switch is in the REFUEL position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. When the mode switch is in the refuel position only one control rod can be withdrawn. The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor is

* The refueling platform frame-mounted, monorail and the service platform fuel-loaded hoist interlocks are required to be OPERABLE only when utilized for in-vessel fuel movements.

subcritical even when the highest worth control rod is fully withdrawn. The combination of refueling interlocks for control rods 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.

3.10 BASES (Cont'd)

3.10.A (Cont'd)

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

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. Each SRM (FLC) is not required to read ≥ 3 cps until after four fuel assemblies have been loaded adjacent to the SRM (FLC) if no other fuel assemblies are in the associated core quadrant. These four locations are adjacent to the SRM dry tube. When utilizing FLCs, the FLCs will be located such that the required count rate is achieved without exceeding the SRM upscale setpoint. With four fuel assemblies or fewer loaded around each SRM, even with a control rod withdrawn, the configuration will not be critical.

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 once per 12 hours verification of the SRM count rate and signal-to-noise ratio 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)

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 temperatures 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.

3.10.F Spent Fuel Cask Handling - Refueling Floor

Although single failure protection has been provided in the design of the 125-ton hoist drum shaft, wire ropes, hook and lower block assembly on the reactor building crane, the limiting of lift height of a spent fuel cask controls the amount of energy available in a dropped cask accident when the cask is over the refueling floor.

An analysis has been made which shows that the floor and support members in the area of cask entry into the decontamination facility can satisfactorily sustain a dropped cask from a height of three feet.

The yoke safety links provide single failure protection for the hook and lower block assembly and limit cask rotation. Cask rotation is necessary for decontamination and the safety links are removed during decontamination.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all required refueling equipment interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform, and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its function.

B. Core Monitoring

Requiring the SRMs to be functionally tested prior to any CORE ALTERATION assures that the SRMs will be OPERABLE at the start of that alteration. The once per 12 hours verification of the SRM count rate and signal-to-noise ratio ensures their continued OPERABILITY.

REFERENCES

1. Fuel Pool Cooling and Cleanup System (BFNP FSAR Subsection 10.5)

2. Spent Fuel Storage (BFNP FSAR Subsection 10.3)

6.2.2 (Cont.)

d. Two licensed reactor operators shall be in the control room during any cold startups, while shutting down the reactor, and during recovery from unit trip. In addition, a person holding a senior operator license shall be in the control room for that unit whenever it is in an operational mode other than cold shutdown or refueling.

e. A Health Physics Technician* shall be present at the facility at all times when there is fuel in the reactor.

f. Either a licensed SRO or licensed SRO limited to fuel handling who has no concurrent responsibilities during this operation shall be present during fuel handling and shall directly supervise all CORE ALTERATIONS.

g. Deleted.

*The Health Physics Technician may be absent from the facility for a period of time not to exceed 2 hours provided immediate action is taken to fill the required position.

Table 6.2.A
Minimum Shift Crew Requirements[b]

Position	Units in Operation 0	1	2[d]	3	Type of License
Senior Operator[a]	1	1	1	1	SRO
Senior Operator	0	1	2	2	SRO
Licensed Operators	3	3	3	3	RO or SRO
Additional Licensed Operators[c]	0	1	2	2	RO or SRO
Assistant Unit Operators (AUO)	4	4	5	5	None
Shift Technical Advisor (STA)	0	1	1	1	None
Health Physics Technician	1	1	1	1	None

Note for Table 6.2.A

a. A senior operator will be assigned responsibility for overall plant operation at all times there is fuel in any unit.

b. Except for the senior operator discussed in note "a", the shift crew composition may be one less than the minimum requirements of Table 6.2.A for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.A. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

c. One of the Additional Licensed Operators must be assigned to each control room with an operating unit.

d. The number of required licensed personnel, when the operating units are controlled from a common control room, are two senior operators and four operators.

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

ENCLOSURE 4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 194 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 209 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 INTRODUCTION

By letter dated October 9, 1992, as supplemented on March 31, 1993, the Tennessee Valley Authority (TVA, the licensee) submitted proposed Technical Specification (TS) amendments for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3. The proposed amendments consist of revised surveillance requirements for certain refueling equipment interlocks, revised core reactivity monitoring requirements for refueling, and provided new requirements for dedicated personnel assigned to refueling operations. The supplemental letter dated March 31, 1993, provided clarifying information that did not alter the staff's initial proposed determination of no significant hazards consideration.

2.0 EVALUATION

A new TS Definition 1.S, CORE ALTERATION, is proposed. The new definition states that:

> "CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components [such as control rods], or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel."

The new definition specifically states that movement of source range monitors (SRMs), intermediate range monitors, traversing in-core probes, or special movable detectors, including under-vessel replacement of these components, is not considered a core alteration. Though these detectors may contain fissionable material, their effect on core reactivity is insignificant because the quantities are very small. Furthermore, movement of these detectors does not change core geometry and increase core reactivity. Therefore, the proposed changes are acceptable.

The existing BFN technical specifications provide for the surveillance testing of all refueling equipment interlocks, whether they are scheduled to be used in a certain operation or not.

TS 3.10.A.1 and 3.10.A.6 specify operability requirements for equipment interlocks required for the refueling operations. Refueling operations include the movement of spent and fresh fuel between the core and the fuel storage pool when the vessel head is removed. The refueling interlocks prohibit faulty fuel movements which could make the reactor critical. This is accomplished by restricting control rod movements and by restricting the movement of refueling equipment. The frame-mounted, the monorail-mounted and the service platform hoists are equipped with load cells to provide a signal to the refueling interlock circuitry and must be tested prior to use.

However, not all components are used in every refueling. The proposed change allows for testing the sensing interlocks of just the equipment that will actually be used for fuel handling. The proposed TS 3.10.A/4.10.A provide for surveillance testing of the refueling equipment required to be used in a certain refueling operation. Equipment not scheduled to be used for a specific refueling operation will still be tested but only at the beginning of the operation. It will not be tested every seven days thereafter, as required by the existing specification, unless the licensee elects to use the equipment. No credit is taken for equipment not used in the refueling operation to mitigate any design basis accident in the BFN Final Safety Analysis Report, Sections 14.6.4 and 7.6.4. Therefore, the proposed changes to TS 3.10.A/4.10.A are acceptable.

The current NRC required core monitoring during refueling operations consists of two source range monitors (SRMs). One is located in the quadrant where core operations are performed, and the other located in an adjacent quadrant. In addition, each channel is required to be demonstrated operable every 12 hours.

The proposed modification of TS 3.10.B requires a minimum of two operable SRMs during any fueling operation (or, alternatively, operable fuel loading chambers (FLC)); and requires operable SRMs in all fueled regions, the quadrant of core alterations and the region(s) or quadrants adjacent to it. A fueled region is described as any set of contiguous fuel cells containing one or more fuel assemblies. An operable SRM is considered to be included in the fueled region when one or more of the four locations adjacent to the SRM dry tube contains a fuel assembly. (An FLC may be considered to be in the fueled region even though its actual location is outside the fueled region if it is positioned such that it is monitoring the fuel assemblies in its associated core quadrant.) Therefore, a minimum of two and a maximum of four operable SRMs are required. The corresponding surveillance requirements in 4.10.B.1 define a minimum surveillance frequency of 12 hours. The minimum count rate is specified to be greater than or equal to 3 counts per second, and the signal-to-noise ratio must be greater than 3:1. The proposed technical specifications are more conservative from the existing in that they provide more operable SRMs during the fueling operation. Therefore, the proposed changes are acceptable.

Finally, changes are proposed to TS 6.2.2.f, revising requirements to have a licensed Senior Reactor Operator (SRO), or a licensed SRO limited to fuel handling, in direct supervision of all core alterations. The current specification requires this SRO to be present on the refueling floor while core alterations are performed. However, certain operations, such as control rod movement, may be better directed if the SRO is in another plant location, such as the control room. The revised specification provides this flexibility, but does not diminish the SRO requirements or responsibilities for fuel handling supervision, and is therefore acceptable.

3.0 SUMMARY AND CONCLUSIONS

We reviewed the submitted information regarding the proposed technical specification changes and we find them acceptable based on our assessment that they provide a conservative basis for core alterations and fuel movement.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments revise requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes a surveillance requirement. The amendments also relate to changes in recordkeeping, reporting, or administrative procedures or requirements. The NRC staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment changes involve no significant hazards consideration, and there has been no public comment on such finding (57 FR 55592). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environment assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will no be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: L. Lois and J. Williams
Date: April 9, 1993