OCTOBER 1 1 1979

Distribution
Docket
ORB #3
NRR Reading
Local PDR
NRC PDR
DEisenhut
RVollmer
JMiller
BGrimes
LShao
WGammill
TIppolito
SSheppard
RClark
Atty, OELD
OI&E (5)
BJones (12)
BScharf (10)
STSG
CMiles
RDiggs
HDenton
TERA
JRBuchanan

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

Mr. Hugh G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

The Commission has issued the enclosed Amendment Nos. 53, 48 and 25 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments are in response to your letter of July 20, 1979 (TVA BFNP TS 126) and to our generic letter to you of August 25, 1977 on respiratory protective equipment.

These amendments change the Technical Specifications to (1) allow the count rate in the Source Range Monitor channels to drop below 3 counts per second when the entire reactor core is being removed or replaced and (2) delete Sections 6.3.D.3, 6.3.D.4, 6.3.D.5 and Table 6.3.A on respiratory protective equipment.

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

Sincerely,
Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 53 to DPR-33
2. Amendment No. 48 to DPR-52
3. Amendment No. 25 to DPR-68
4. Safety Evaluation
5. Notice

7910260 O46 P

No legal objection to Amendment + notice. SER not reviewed

CCP
CP 2

cc w/enclosures:
See next page

OFFICE	ORB #3	ORB #3	AD:OR	OELD	ORB #3
SURNAME	SSheppard	RClark:mjf	WGammill	CUTCHIN	TIppolito
DATE	10/2/79	10/2/79	10/2/79	10/5/79	10/9/79

cc:

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U. S. Environmental Protection Agency
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Mr. Robert F. Sullivan
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 20, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 53, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

7910260 021

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 11, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 53

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

304/305	338/339
306/307 *	340/341
310/311	342/343
312/313 *	344/345

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

*There are no marginal lines on pages 306 and 312. The change on page 306 is to move, verbatim, paragraph 3.10.B.2 from the bottom of page 305 to the top of page 306. The change on page 312 is to move the first paragraph of 3.10.C "Spent Fuel Pool Water" verbatim from the bottom of page 311 to the top of page 312.

3.10.A Refueling Interlocks

refueling interlocks shall be operable.

b. A sufficient number of control rods shall be operable so that the core can be made subcritical with the strongest operable control rod fully withdrawn and all other operable control rods fully inserted, or all directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.

c. If maintenance is to be performed on two control rod drives they must be separated by more than two control cells in any direction.

d. An appropriate number of SRM's are available as defined in specification 3.10.B.

6. 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

4.10.A Refueling Interlocks

3. With the mode selection switch in the refuel or shutdown mode, no control rod may be withdrawn until two licensed operators have confirmed that either all fuel has been removed from around that rod or that all control rods in immediately adjacent cells have been fully inserted and electrically disarmed.

Amendment No. 53

3.10.A <u>Refueling Interlocks</u>

being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

B. <u>Core Monitoring</u>

1. During core alterations, except as in 3.10.B.2, two SRM's shall be operable, in or adjacent to any quadrant where fuel or control rods are being moved. For an SRM 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.1 The SRM shall have a minimum of 3 cps with all rods fully inserted in the core, if one or more fuel assemblies are in the core, or,

   b.2 During a full core reload where both irradiated and fresh fuel is being loaded, SRM's (FLC's) may have a count rate of <3 cps provided that the SRM's are response checked at least once every 8 hours with a neutron source until >3 cps can be maintained, and provided also that the core is loaded in a spiral sequence only.

4.10.A <u>Refueling Interlocks</u>

B. <u>Core Monitoring</u>

Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response except as specified in 3.10.B.1.b.2.

2. During a complete core removal, the SRM's shall have an initial minimum count rate of 3 cps prior to fuel removal, with all rods fully inserted and rendered electrically inoperable. The count rate will diminish 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.C <u>Spent Fuel Pool Water</u>

1. Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at a depth of 8 1/2 feet or greater above the top of the spent fuel. A minimum of 6-1/2 feet of water shall be maintained over single irradiated fuel assemblies during transfer and handling operations.

2. Whenever irradiated fuel is in the fuel pool, the pool water temperature shall be $\leq$ 150°F.

3. Fuel pool water shall be maintained within the following limits:

   conductivity $\leq$ 10 umhos/cm @25°C

   chlorides $\leq$ 0.5 ppm

4.10.C <u>Spent Fuel Pool Water</u>

1. Whenever irradiated fuel is stored in the spent fuel pool, the water level and temperature shall be recorded daily

2. A sample of fuel pool water shall be analyzed in accordance with the following specifications:

   a. At least daily for conductivity and chloride ion content.

   b. At least once per 8 hours for conductivity and chloride content when the fuel pool cleanup system is inoperable.

3.10.D Reactor Building Crane

1. The reactor building crane shall be operable:

   a. When a spent fuel cask is handled.

   b. Whenever new or spent fuel is handled with the 5-ton hoist.

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.

4.10.D Reactor Building Crane

1. The following operational checks and inspections shall be performed on the reactor building crane prior to handling of a spent fuel cask and new or spent fuel. (These need not be performed more frequently than quarterly.):

   a. The cab and pendant controls shall be demonstrated to be operable on both the 125-ton hoist and the 5-ton hoist.

   b. A visual inspection shall be made to insure structural integrity of the 125-ton hoist, the 5-ton hoist and cask yoke safety wire ropes.

   c. The overtravel limit switch interlocks, movement speed control and braking operations for the bridge, trolley and hoists, the pendant interlocks, the main-auxiliary hoist operation interlock, and the remote emergency stop shall be functionally tested.

E. Spent Fuel Cask

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.

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. 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 or 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 the core is shut down by a margin of 0.38 percent Δk with the strongest operable control rod fully withdrawn, or that at least 0.38% Δk shutdown margin is available if the remaining control rods have had their directional control valves disarmed. Disarming the directional control valves does not inhibit control rod scram capability.

Specification 3.10.A.6 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

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

B. Core Monitoring

The SRM's 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 SRM's in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored and ensures that startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident.

During a full core reload SRM/FLC (Fuel Loading Chamber) operability will be verified using a portable external source at least once every 8 hours until sufficient fuel has been loaded to maintain 3 cps. A large number of fuel assemblies will not be required to maintain 3 cps. This increased surveillance rate assures proper detector operability until that time.

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 3 cps. All fuel moves during core unloading will reduce reactivity. It is expected that the SRM's will drop below 3 cps 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 all of the fuel has been removed to the spent fuel storage pool, SRM's will no longer be required. Requiring the SRM's to be functionally tested prior to fuel removal assures that the SRM's will be operable at the start of fuel removal. The daily response check of the SRM's ensures their continued operability until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

REFERENCES

1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

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

3.10 BASES

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 5-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 3 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.

6.3 Procedures

A. Detailed written procedures, including applicable check-off lists covering items listed below shall be prepared, approved and adhered to.

1. Normal startup, operation and shutdown of the reactor and of all systems and components involving nuclear safety of the facility.
2. Refueling operations.
3. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.
4. Emergency conditions involving potential or actual release of Radioactivity.
5. Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.
6. Surveillance and testing requirements.
7. Radiation control procedures.
8. Radiological Emergency Plan implementing procedures.
9. Plant security program implementing procedures.
10. Fire protection and prevention procedures.

B. Written procedures pertaining to those items listed above shall be reviewed by PORC and approved by the plant superintendent prior to implementation. Temporary changes to a procedures which do not change the intent of the approved procedure may be made by a member of the plant staff knowledgeable in the area affected by the procedure except that temporary changes to those items listed above except item 5 require the additional approval of a member of the plant staff who holds a Senior Reactor Operator license on the unit affected. Such changes shall be documented and subsequently reviewed by PORC and approved by the plant superintendent.

C. Drills on actions to be taken under emergency conditions involving release of radioactivity are specified in the radiological emergency plan and shall be conducted annually. Annual drills shall also be conducted on the actions to be taken following failures of safety related systems or components.

D. Radiation Control Procedures

Radiation Control Procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20 except in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

1. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1,000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area, and entrance thereto shall be controlled by issuance of a special work permit. Any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose while in the area.

2. Each High Radiation Area in which the intensity of radiation is greater than 1,000 mrem/hr shall be subject to the provisions of (a) above; and, in addition, locked doors shall be provided to prevent unauthorized entry into such areas, and the keys shall be maintained under administrative control of the shift engineer on duty.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 20, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 11, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

   303/304
   305/306*
   311/312*
   339/340
   341/342
   343/344
   345/346

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

*There are no marginal lines on pages 306 and 312. The change on page 306 is to move, verbatim, paragraph 3.10.B.2 from the bottom of page 305 to the top of page 306. The change on page 312 is to move the first paragraph of 3.10.C "Spent Fuel Pool Water" verbatim from the bottom of page 311 to the top of page 312.

3.10.A Refueling Interlocks

3. The fuel grapple hoist load switch shall be set at ≤ 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.

5. A maximum of two non-adjacent control rods may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance, 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

4.10.A Refueling Interlocks

control rods are fully inserted and have had their directional control valves electrically disarmed, it is sufficient to demonstrate that the core is subcritical with a margin of at least 0.38 Δk at any time during the maintenance. A control rod on which maintenance is being performed shall be considered inoperable.

3.10.A Refueling Interlocks

refueling interlocks shall be operable.

b. A sufficient number of control rods shall be operable so that the core can be made subcritical with the strongest operable control rod fully withdrawn and all other operable control rods fully inserted, or all directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.

c. If maintenance is to be performed on two control rod drives they must be separated by more than two control cells in any direction.

d. An appropriate number of SRM's are available as defined in specification 3.10.B.

6. 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

4.10.A Refueling Interlocks

3. With the mode selection switch in the refuel or shutdown mode, no control rod may be withdrawn until two licensed operators have confirmed that either all fuel has been removed from around that rod or that all control rods in immediately adjacent cells have been fully inserted and electrically disarmed.

3.10.A Refueling Interlocks

being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

B. Core Monitoring

1. During core alterations, except as in 3.10.B.2, two SRM's shall be operable, in or adjacent to any quadrant where fuel or control rods are being moved. For an SRM 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.1 The SRM shall have a minimum of 3 cps with all rods fully inserted in the core, if one or more fuel assemblies are in the core, or,

   b.2 During a full core reload where both irradiated and fresh fuel is being loaded, SRM's (FLC's) may have a count rate of <3 cps provided that the SRM's are response checked at least once every 8 hours with a neutron source until >3 cps can be maintained, and provided also that the core is loaded in a spiral sequence only.

4.10.A Refueling Interlocks

B. Core Monitoring

Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response except as specified in 3.10.B.1.b.2.

2. During a complete core removal, the SRM's shall have an initial minimum count rate of 3 cps prior to fuel removal, with all rods fully inserted and rendered electrically inoperable. The count rate will diminish 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.C Spent Fuel Pool Water

1. Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at a depth of 8 1/2 feet or greater above the top of the spent fuel. A minimum of 6-1/2 feet of water shall be maintained over single irradiated fuel assemblies during transfer and handling operations.

2. Whenever irradiated fuel is in the fuel pool, the pool water temperature shall be $\leq 150^{o}F$.

3. Fuel pool water shall be maintained within the following limits:

   conductivity $\leq$ 10 umhos/cm @25°C

   chlorides $\leq$ 0.5 ppm

4.10.C Spent Fuel Pool Water

1. Whenever irradiated fuel is stored in the spent fuel pool, the water level and temperature shall be recorded daily

2. A sample of fuel pool water shall be analyzed in accordance with the following specifications:

   a. At least daily for conductivity and chloride ion content.

   b. At least once per 8 hours for conductivity and chloride content when the fuel pool cleanup system is inoperable.

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

B. Core Monitoring

The SRM's 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 SRM's in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored and ensures that startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident.

During a full core reload SRM/FLC (Fuel Loading Chamber) operability will be verified using a portable external source at least once every 8 hours until sufficient fuel has been loaded to maintain 3 cps. A large number of fuel assemblies will not be required to maintain 3 cps. This increased surveillance rate assures proper detector operability until that time.

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 3 cps. All fuel moves during core unloading will reduce reactivity. It is expected that the SRM's will drop below 3 cps 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 all of the fuel has been removed to the spent fuel storage pool, SRM's will no longer be required. Requiring the SRM's to be functionally tested prior to fuel removal assures that the SRM's will be operable at the start of fuel removal. The daily response check of the SRM's ensures their continued operability until the count rate diminishes due to fuel removal. Control rods in cells from which all fuel has been removed and which are outside the periphery of the then existing fuel matrix may be armed electrically and moved for maintenance purposes during full core removal, provided all rods that control fuel are fully inserted and electrically disarmed.

REFERENCES

1. Neutron Monitoring System (BFNP FSAR Subsection 7.5)

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

3.10 BASES

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 5-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.

C. Drills on actions to be taken under emergency conditions involving release of radioactivity are specified in the radiological emergency plan and shall be conducted annually. Annual drills shall also be conducted on the actions to be taken following failures of safety related systems or components.

D. Radiation Control Procedures

Radiation Control Procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20 except in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

1. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1,000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area, and entrance thereto shall be controlled by issuance of a special work permit. Any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose while in the area.

2. Each High Radiation Area in which the intensity of radiation is greater than 1,000 mrem/hr shall be subject to the provisions of (a) above; and, in addition, locked doors shall be provided to prevent unauthorized entry into such areas, and the keys shall be maintained under administrative control of the shift engineer on duty.

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6.4 Actions to be Taken in the Event of a Reportable Occurrence in Plant Operation (Ref. Section 6.7)

A. Any reportable occurrence shall be promptly reported to the Chief, Nuclear Generation Branch and shall be promptly reviewed by PORC. This committee shall prepare a separate report for each reportable occurrence. This report shall include an evaluation of the cause of the occurrence and recommendations for appropriate action to prevent or reduce the probability of a repetition of the occurrence.

B. Copies of all such reports shall be submitted to the Chief, Nuclear Generation Branch, the Manager of Power, the Division of Power Resource Planning, and the Chairman of the NSRB for their review.

C. The plant superintendent shall notify the NRC as specified in Specification 6.7 of the circumstances of any reportable occurrence.

6.5 Action to be Taken in the Event a Safety Limit is Exceeded

If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC. A prompt report shall be made to the Chief, Nuclear Generation Branch and the Chairman of the NSRB. A complete analysis of the circumstances leading up to and resulting from the situation, together with recommendations to prevent a recurrence, shall be prepared by the PORC. This report shall be submitted to the Chief, Nuclear Generation Branch, the Manager of Power, the Division of Power Resource Planning, and the NSRB. Notification of such occurrences will be made to the NRC by the plant superintendent within 24 hours.

6.6 Station Operating Records

A. Records and/or logs shall be kept in a manner convenient for review as indicated below:

1. All normal plant operation including such items as power level, fuel exposure, and shutdowns

2. Principal maintenance activities

3. Reportable occurrences

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 20, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 25, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 11, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

334	370
336	371
337*	372
342	373
343*	374
369	375

2. Marginal lines indicate revised area.

* There are no marginal lines on pages 337 and 343. The change on page 337 is to move, verbatim, paragraph 3.10.B.2 from the bottom of page 336 to the top of page 337. The change on page 343 is to move two references, verbatim, from the bottom of page 342 to the top of page 343.

3.10 CORE ALTERATIONS

b. A sufficient number of control rods shall be operable so that the core can be made subcritical with the strongest operable control rod fully withdrawn and all other operable control rods fully inserted, or all directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.

c. If maintenance is to be performed on two control rod drives they must be separated by more than two control cells in any direction.

d. An appropriate number of SRM's are available as defined in specification 3.10.B.

4.10 CORE ALTERATIONS

3.10 CORE ALTERATIONS

B. Core Monitoring

1. During core alterations, except as in 3.10.B.2, two SRM's shall be operable, in or adjacent to any quadrant where fuel or control rods are being moved. For an SRM 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.1 The SRM shall have a minimum of 3 cps with all rods fully inserted in the core, if one or more fuel assemblies are in the core, or

b.2 During a full core reload where both irradiated and fresh fuel is being loaded, SRM's (FLC's) may have a count rate of <3 cps provided that the SRM's are response checked at least once every 8 hours with a neutron source unitl >3 cps can be maintained, and provided also that the core is loaded in a spiral sequence only.

4.10 CORE ALTERATIONS

B. Core Monitoring

Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response except as specified in 3.10.B.1.b.2.

2. During a complete core removal, the SRM's shall have an initial minimum count rate of 3 cps prior to fuel removal, with all rods fully inserted and rendered electrically inoperable. The count rate will diminish 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.C Spent Fuel Pool Water

1. Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at a depth of 8 1/2 feet or greater above the top of the spent fuel. A minimum of 6-1/2 feet of water shall be maintained over single irradiated fuel assemblies during transfer and handling operations.

2. Whenever irradiated fuel is in the fuel pool, the pool water temperature shall be $\leq 150^{\circ}F$.

3. Fuel pool water shall be maintained within the following limits:

   conductivity $\leq$ 10 umhos/cm @25°C

   chlorides $\leq$ 0.5 ppm

4.10.C Spent Fuel Pool Water

1. Whenever irradiated fuel is stored in the spent fuel pool, the water level and temperature shall be recorded daily

2. A sample of fuel pool water shall be analyzed in accordance with the following specifications:

   a. At least daily for conductivity and chloride ion content.

   b. At least once per 8 hours for conductivity and chloride content when the fuel pool cleanup system is inoperable.

provides primary reactivity control for the fuel assemblies in the cell associated with that control rod.

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

REFERENCES

1. Refueling interlocks (BFNP FSAR Subsection 7.6)

B. Core Monitoring

The SRM's 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 SRM's in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored and insures that startup is conducted only if the source range flux level is above the minimum assumed in the control rod drop accident.

During a full core reload SRM/FLC (Fuel Loading Chamber) operability will be verified using a portable external source at least once every 8 hours until sufficient fuel has been loaded to maintain 3 cps. A large number of fuel assemblies will not be required to maintain 3 cps. This increased surveillance rate assures proper detector operability until that time.

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 3 cps. All fuel moves during core unloading will reduce reactivity. It is expected that the SRM's will drop below 3 cps 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 all of the fuel has been removed to the spent fuel storage pool, SRM's will no longer be required. Requiring the SRM's to be functionally tested prior to fuel removal assures that the SRM's will be operable at the start of fuel removal. The daily response check of the SRM's 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)

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.

6.0 ADMINISTRATIVE CONTROLS

C. Drills on actions to be taken under emergency conditions involving release of radioactivity are specified in the radiological emergency plan and shall be conducted annually. Annual drills shall also be conducted on the actions to be taken following failures of safety related systems or components.

D. Radiation Control Procedures

Radiation Control Procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20 except in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

1. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1,000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area, and entrance thereto shall be controlled by issuance of a special work permit. Any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose while in the area.

2. Each High Radiation Area in which the intensity of radiation is greater than 1,000 mrem/hr shall be subject to the provisions of (a) above; and, in addition, locked doors shall be provided to prevent unauthorized entry into such areas, and the keys shall be maintained under administrative control of the shift engineer on duty.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 Introduction

1.1 Count Rate Requirements for SRMs

By letter dated July 20, 1979 (TVA BFNP TS 126), the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating Licenses Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. The proposed amendments and revised Technical Specifications would (1) allow the count rate in the Source Range Monitor (SRM) channels to drop below 3 counts per second (cps) when the entire reactor core is being removed or replaced and (2) would correct a typographical error in Section 3.10.A.4.d. The present Technical Specifications require that a count rate of at least 3 cps be maintained whenever one or more fuel assemblies are present in the core. With respect to the second item, the limiting condition for operation (LCO) in Section 3.10.A.5.d reads: "An appropriate number of SRMs are available as defined in specification 3.10.A"; this should read "3.10.B", since the latter is the section on core monitoring which addresses the SRM requirements.

1.2 Respiratory Protection Program

On August 25, 1977, the Commission issued a generic letter addressed to the licensee with respect to the respiratory protection program described in Sections 6.3.D.3, 6.3.D.4, 6.3.D.5 and Table 6.3.A of the Technical Specifications for each of the three Browns Ferry units. The letter called attention to the fact that on November 29, 1976, the Commission published in the FEDERAL REGISTER an amended Section 20.103 of 10 CFR 20, which became effective on December 29, 1976. One effect of this revision is that in order to receive credit for limiting the inhalation of airborne radioactive material, respiratory protective equipment must be used as stipulated in Regulatory Guide 8.15. Another requirement of the amended regulation is that licensees authorized to make allowance for use of respiratory protective equipment prior to December 29, 1976, must bring the use of their respiratory protective equipment into conformance with Regulatory Guide 8.15 by December 29, 1977.

The Browns Ferry Technical Specifications anticipated the above Amendment to Section 20.103 of 10 CFR 20; section 6.3.D.5 states:

> "5. These specifications with respect to the provision of 20.103 shall be superseded by adoption of proposed changes to 10 CFR 20, section 20.103, which would make this specification unnecessary."

In our letter of August 25, 1977, we advised TVA that "In view of the provisions of Section 6.3.D of your Technical Specifications, which require conformance with 10 CFR 20, the fact that Section 20.103 no longer requires specific authorization to employ respiratory protective equipment, and the revocation provisions of subsection 6.3.D.5, we conclude that the necessary amendment to your facility's Technical Specifications can be effected by merely deleting Sections 6.3.D.3, 6.3.D.4, 6.3.D.5 and Table 6.3.A."

In the letter, we also advised TVA that "Based on the revocation provision of your current specification on respiratory protection and in the absence of prior written objection from you, we will include deletion of this specification in an amendment of your Technical Specifications approved after December 28, 1977. No response to this letter is required".

This amendment will delete Sections 6.3.D.4, 6.3.D.5 and Table 6.3.A in accordance with our letter of August 25, 1977. There is no safety significance since these sections are in effect revoked by 10 CFR 20.103.

2.0 Discussion

During any core alteration, and especially during core loading, it is necessary to monitor flux levels. In this manner, even in the highly unlikely event of multiple errors, there is reasonable assurance that any approach to criticality would be detected in time to halt operations.

The minimum count rate requirement in the Technical Specifications accomplishes three safety functions: (1) it assures the presence of some neutron in the core, (2) it provides assurance that the analog portion of the SRM channels is operable, and (3) it provides assurance that the SRM detectors are close enough to the array of fuel assemblies to monitor core flux levels.

Unloading and reloading of the entire core leads to some difficulty with this minimum count rate requirement. When only a small number of assemblies are present within the core, the SRM count rate will drop below the minimum due to the small number of neutrons being produced, and due to attenuation of these neutrons in the water (and control blades) separating the fuel from the SRM detectors. Past practice has been to connect temporary "dunking" chambers to the SRM channels in place of the normal detectors, and to locate these detectors near the fuel.

Besides being operationally inconvenient, dunking chambers suffer from signal variations due to their lack of fixed geometry. Moreover, the use of dunking chambers increases the risk of loose objects being dropped into the vessel.

3.0 Evaluation

3.1 Minimum Flux in the Core

A multiplying medium with no neutrons present forms the basis for an accident scenario in which reactivity is gradually but inadvertently added until the medium is highly supercritical. No neutron flux will be evident since there are no neutrons present to be multiplied. The introduction of some neutrons at this point would cause the core to undergo a sudden power burst, rather than a gradual startup, with no warning from the nuclear instrumentation.

This scenario is of great concern when loading fresh fuel, but is of lesser concern for exposed fuel. Exposed fuel continuously produces neutrons by spontaneous fission of certain plutonium isotopes, photofission, and some delayed neutron emission. This neutron production in exposed fuel is normally great enough to meet the 3 cps minimum for a full core after a refueling outage with the lumped neutron sources removed.

Thus, there is assurance that a minimum flux level will be present as long as some exposed fuel is present. We therefore find the proposed amendment to be acceptable from the point of view of minimum flux provided the words "full core reload" in Specification 3.10.B.1.b.2 are interpreted to mean "reload of fuel which has previously accumulated exposure in the reactor." We do not find the amendment to be applicable to the loading of a new core containing only fresh fuel. Such a loading must use lumped neutron sources and dunking chambers to meet the normal 3 cps minimum count rate. On September 19, 1979, TVA proposed alternative wording to Section 3.10.B.2 of the Technical Specifications specifying that the minimum count rate requirement of <3cps only applies when both irradiated and fresh fuel is being loaded; this change satisfies the staff's concern above and is acceptable.

3.2 SRM Operability

The amended Specifications 3.10.B.1.b.2 and 4.10.B will require a functional check of the SRM channels by means of a neutron source prior to beginning core alterations and at least every eight hours thereafter. The required interval for other types of alterations is usually once per day. This would be sufficient for core unloading and reloading except that the more extensive fuel handling operations involved load to a slightly greater possibility of SRM failure. We agree that a tripled test frequency is sufficient to cover this, and therefore find the eight hour interval to be acceptable.

3.3 Flux Attenuation

The four SRM detectors are located, one per quadrant, roughly half a core radius from the center. Although these are incore detectors and thus very sensitive when the reactor is fully loaded, they lose some of their effectiveness when the reactor is partially defueled and the detectors are located some distance from the array of remaining fuel.

GE's spent fuel pool studies have shown(1) that 16 or more fuel assemblies (i.e., four or more control cells) must be loaded together before criticality is possible. In spiral (and most other) loading sequences in the Browns Ferry cores, an array containing four or more control cells will be at most two control cells (i.e., about two feet) away from an SRM detector. We have previously examined the sensitivity loss in such a case on another docket, (2) and found it to be at most one decade of sensitivity (i.e., about one fifth of the SRM's logarithmic scale). We find this to be acceptable.

However, there are areas near the 90° and 180° sides of the Browns Ferry cores where it is possible to load fuel at a still greater distance from the nearest detector. In the absence of quantitative justification by the licensee, we cannot find this amendment acceptable for all possible loading sequences. Therefore, we find this amendment to be acceptable only for spiral unloading/reloading sequences, which we understand to be the only sequences the licensee plans to actually use. By "spiral sequences," we mean any sequence in which the central control cell is last unloaded and first reloaded, all fueled locations are contiguous, and no imbedded cavities or major peripheral concavities are permitted. On September 19, 1979, TVA proposed alternative working to Section 3.10.B.2 which specifies that the less than 3 cps only applies when the core is loaded in a spiral sequence; this satisfies the staff's concern and is acceptable.

4.0 Environmental Considerations

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

5.0 Conclusion

We have concluded that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 11, 1979

REFERENCES

1. General Electric Standard Safety Analysis Report, 251-GESSAR, Section 4.3.2.7, p. 4.3-27.

2. "Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 27 to Facility Operating License No. DPR-63," Docket No. 50-220, enclosed with letter, Thomas A. Ippolito (NRC) to Donald P. Dise (Niagara Mohawk Power Corporation), dated March 2, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-259, 50-260 AND 50-296

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 53 to Facility Operating License No. DPR-33, Amendment No. 48 to Facility Operating License No. DPR-52 and Amendment No. 25 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

These amendments change the Technical Specifications to (1) allow the count rate in the Source Range Monitor channels to drop below 3 counts per second when the entire reactor core is being removed or replaced and (2) delete the sections on respiratory protective equipment which are no longer applicable due to the Commission's amendment of Section 20.103 of 10 CFR 20.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated July 20, 1979, (2) Amendment No. 53 to License No. DPR-33, Amendment No. 48 to License No. DPR-52, and Amendment No. 25 to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 11th day of October 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors