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August 6, 1981

RECEIVED AUG 18 1981 U.S. NUCLEAR REGULATORY COMMISSION

Docket No. 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 No. 74 , 71 and 46 to Facility License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3. These amendments are in response to your letter of April 28, 1981 (TVA BFNP TS 159).

The amendments change the Technical Specifications to clarify radiation monitoring requirements, remove a redundant limit on spent fuel storage and modify the requirements on the residual heat removal service water pumps.

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

Sincerely,

ORIGINAL SIGNED BY

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

Enclosures:
1. Amendment No. 74 to DPR-33
2. Amendment No. 71 to DPR-52
3. Amendment No. 46 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures: See next page

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no legal objections

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DATE	7/31/81	7/31/81	8/5/81	8/3/81	8/3/81		

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

Mr. Hugh G. Parris

cc:

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. DPR-33

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

   A. The application for amendments by Tennessee Valley Authority (the licensee), dated April 28, 1981 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. 74 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachments:
Changes to the Technical
Specifications

Date of Issuance: August 6, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 74

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Review Appendix as Follows:

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

   7/8
   152
   330/331
   338/339

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

3. Add the following new page:

   152a

10. Logic - A logic is an arrangement of relays, contacts, and other components that produces a decision output.

    (a) Initiating - A logic that receive signals from channels and produces decision outputs to the actuation logic.

    (b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

W. Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

X. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

Y. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

1. FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specifications

A. Reactor Pressure > 800 psia and Core Flow > 10% of Rated.

When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$$S \leq (0.66W + 54\%)$$

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)

2. During reactor power operation, RHRSW pumps must be operable and assigned to service as indicated in Table 3.5-1 for the specified time limits.

3. During power operation, both RHRSW pumps D1 and D2 normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be operable; except as specified in 3.5.C.4 and 3.5.C.5 below.

4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)

2. a. If no more than two RHRSW pumps are inoperable, increased surveillance is not required.

b. When three RHRSW pumps are inoperable, the remaining pumps, associated essential control valves, and associated diesel generators shall be operated weekly.

c. When four RHRSW pumps are inoperable, the remaining pumps, associated essential control valves, and associated diesel generators shall be operated daily.

3. Routine surveillance for these pumps is specified in 4.5.C.1.

TABLE 3.5-1

Time Limit (Days)	Minimum Service Assignment RHRSW	EECW (2)
Indefinite	7 (4)	3 (1)
30	7 or 6 (3)(4)	2 (1) or 3 (3)
7	6 (4)	2 (1)

(1) At least one operable pump must be assigned to each header.

(2) Only automatically starting pumps may be assigned to EECW header service.

(3) Nine pumps must be operable. Either configuration is acceptable: 7 and 2 or 6 and 3.

(4) Requirements may be reduced by two for each unit with fuel unloaded.

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry unit 1 is located at Browns Ferry Nuclear Plant site on property owned by the United States and in custody of the TVA. The site shall consist of approximately 840 acres on the north shore of Wheeler Lake at Tennessee River Mile 294 in Limestone County, Alabama. The minimum distance from the outside of the secondary containment building to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 4,000 feet.

5.2 REACTOR

A. The reactor core may contain 764 fuel assemblies consisting of 7x7 assemblies having 49 fuel rods each, 8x8 assemblies having 63 fuel rods each, and 8x8R (and P8x8R) assemblies having 62 fuel rods each.

B. The reactor core shall contain 185 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70 percent of theoretical density.

5.3 REACTOR VESSEL

The reactor vessel shall be as described in Table 4.2-2 of the FSAR. The applicable design codes shall be as described in Table 4.2-1 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters for the primary containment shall be as given in Table 5.2-1 of the FSAR. The applicable design codes shall be as described in Section 5.2 of the FSAR.

B. The secondary containment shall be as described in Section 5.3 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with the standards set forth in Section 5.2.3.4 of the FSAR.

5.5 FUEL STORAGE

A. The arrangement of fuel in the new-fuel storage facility shall be such that k_{eff}, for dry conditions, is less than 0.90 and flooded is less than 0.95 (Section 10.2 of FSAR).

FEB 25 1980

5.0 MAJOR DESIGN FEATURES (Continued)

B. The k_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

C. Loads greater than 1000 pounds shall not be carried over spent fuel assemblies stored in the spent fuel pool.

5.6 SEISMIC DESIGN

The station class I structures and systems have been designed to withstand a design basis earthquake with ground acceleration of 0.2g. The operational basis earthquake used in the plant design assumed a ground acceleration of 0.1g (see Section 2.5 of the FSAR).

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) of 10 CFR 20:

1. Each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit[a]. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

   a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

   c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Special Work Permit.

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 (1) 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.

[a] Health Physics personnel, or personnel escorted by Health Physics personnel, in accordance with approved emergency procedures, shall be exempt from the SWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. DPR-52

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

   A. The application for amendments by Tennessee Valley Authority (the licensee) dated April 28, 1981 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. 71, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to Technical
Specifications

Date of Issuance: August 6, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 71

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.

   7/8

   152

   331/332

   339/340

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

3. Add the following new page:

   152a

1.0 DEFINITIONS (Cont'd)

10. Logic - A logic is an arrangement of relays, contacts, and other components that produces a decision output.

(a) Initiating - A logic that receive signals from channels and produces decision outputs to the actuation logic.

(b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

W. Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

X. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

Y. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

1. FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specifications

A. Reactor Pressure > 800 psia and Core Flow > 10% of Rated.

When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$$S \leq (0.66W + 54\%)$$

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

Amendment No. 35, 46

MAY 25 1979

3.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)

2. During reactor power operation, RHRSW pumps must be operable and assigned to service as indicated in Table 3.5-1 for the specified time limits.

3. During power operation, both RHRSW pumps D1 and D2 normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be operable; except as specified in 3.5.C.4 and 3.5.C.5 below.

4.5.C RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS) (Continued)

2. a. If no more than two RHRSW pumps are inoperable, increased surveillance is not required.

   b. When three RHRSW pumps are inoperable, the remaining pumps, associated essential control valves, and associated diesel generators shall be operated weekly.

   c. When four RHRSW pumps are inoperable, the remaining pumps, associated essential control valves, and associated diesel generators shall be operated daily.

3. Routine surveillance for these pumps is specified in 4.5.C.1.

TABLE 3.5-1

Time Limit (Days)	Minimum Service Assignment RHRSW	EECW [2]
Indefinite	7 [4]	3 [1]
30	7 or 6 [3][4]	2 [1] or 3 [3]
7	6 [4]	2 [1]

(1) At least one operable pump must be assigned to each header.

(2) Only automatically starting pumps may be assigned to EECW header service.

(3) Nine pumps must be operable. Either configuration is acceptable: 7 and 2 or 6 and 3.

(4) Requirements may be reduced by two for each unit with fuel unloaded.

5.0 MAJOR DESIGN FEATURES (Continued)

B. The k_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

C. Loads greater than 1000 pounds shall not be carried over spent fuel assemblies stored in the spent fuel pool.

5.6 SEISMIC DESIGN

The station class I structures and systems have been designed to withstand a design basis earthquake with ground acceleration of 0.2g. The operational basis earthquake used in the plant design assumed a ground acceleration of 0.1g (see Section 2.5 of the FSAR).

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

A. The plant superintendent has on-site responsibility for the safe operation of the facility and shall report to the Chief, Nuclear Generation Branch. In the absence of the plant superintendent, the assistant superintendnet will assume his responsibilities.

B. The portion of TVA management which relates to the operation of the plant is shown in Figure 6.1-1.

C. The functional organization for the operation of the station shall be as shown in Figure 6.1-2.

D. Shift manning requirements shall, as a minimum, be as described in section 6.8.

E. Qualifications of the Browns Ferry Nuclear Plant management and operating staff shall meet the minimum acceptable levels as described in ANSI - N18.1, Selection and Training of Nuclear Power Plant Personnel, dated March 8, 1971. The qualifications of the Health Physics Supervisor will meet or exceed the minimum acceptable levels as described in Regulatory Guide 1.8, Revision 1, dated Sept. 1975.

F. Retraining and replacement training of station personnel shall be in accordance with ANSI - N18.1, Selection and Training of Nuclear Power Plant Personnel, dated March 8, 1971. The minimum frequency of the retraining program shall be every two years.

G. An Industrial Security Program shall be maintained for the life of the plant.

H. Responsibilities of a post-fire overall restoration coordinator will consist of duties as described in section 6.9.

I. The Safety Engineer shall have the following qualifications:

a. Must have a sound understanding and thorough technical knowledge of safety and fire protection practices, procedures, standards, and other codes relating to electrical utility operations. Must be able to read and understand engineering drawings. Must possess an analytical ability for problem solving and data analysis. Must be able to communicate well both orally and in writing and must be able to write investigative reports and prepare written procedures. Must have the ability to secure the cooperation of management, employees and groups in the implementation of safety programs. Must be able to conduct safety presentations for supervisors and employees.

b. Should have experience in safety engineering work at this level or have 3 years experience in safety and/or fire protection engineering. It is desirable that the incumbent be a graduate of an accredited college or university with a degree in inductrial, mechanical, electrical, or safety engineering or fire protection engineering.

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) of 10 CFR 20:

1. Each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit[a]. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

   a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

   c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Special Work Permit.

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 (1) 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.

[a] Health Physics personnel, or personnel escorted by Health Physics personnel, in accordance with approved emergency procedures, shall be exempt from the SWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

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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. 46
License No. DPR-68

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

   A. The application for amendments by Tennessee Valley Authority (the licensee) dated April 28, 1981 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. 46, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 6, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 46

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:

   7
   156
   361
   369

2. Add the following new page:

   156a

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

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 conincident 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 Acutation - Simulated automatic acutation means applying a simulated signal to the sensor to actuate the circuit in question.

10. Logic - A logic is an arrangement of relays, contacts, and other components that produces a decision output.

    (a) Initiating - A logic that receives signals from channels and produces decision outputs to the actuation logic.

    (b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

W. Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

X. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

Y. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

2. During reactor power operation, RHRSW pumps must be operable and assigned to service as indicated in Table 3.5-1 for the specified time limits.

3. During power operation, both RHRSW pumps B1 and B2 normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be operable; except as specified in 3.5.C.4 and 3.5.C.5 below.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

2. a. If no more than two RHRSW pumps are inoperable, increased surveillance is not required.

   b. When three RHRSW pumps are inoperable, the remaining pumps, associated essential control valves, and associated diesel generators shall be operated weekly.

   c. When four RHRSW pumps are inoperable, the remaining pumps, associated essential control valves, and associated diesel generators shall be operated daily.

3. Routine surveillance for these pumps is specified in 4.5.C.1.

TABLE 3.5-1

Time Limit (Days)	Minimum Service Assignment RHRSW	Minimum Service Assignment EECW [2]
Indefinite	7 [4]	3 [1]
30	7 or 6 [3][4]	2 [1] or 3 [3]
7	6 [4]	2 [1]

(1) At least one operable pump must be assigned to each header.

(2) Only automatically starting pumps may be assigned to EECW header service.

(3) Nine pumps must be operable. Either configuration is acceptable: 7 and 2 or 6 and 3.

(4) Requirements may be reduced by two for each unit with fuel unloaded.

is less than 0.90 and flooded is less than 0.95 (Section 10.2 of FSAR).

B. The k_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

C. Loads greater than 1000 pounds shall not be carried over spent fuel assemblies stored in the spent fuel pool.

5.6 SEISMIC DESIGN

The station class I structures and systems have been designed to withstand a design basis earthquake with ground acceleration of 0.2g. The operational basis earthquake used in the plant design assumed a ground acceleration of 0.1g (see Section 2.5 of the FSAR).

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) of 10 CFR 20:

1. Each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit[a]. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

   a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

   c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Special Work Permit.

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 (1) 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.

[a] Health Physics personnel, or personnel escorted by Health Physics personnel, in accordance with approved emergency procedures, shall be exempt from the SWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 Introduction

By letter dated April 28, 1981 (TVA BFNP TS 159), the Tennessee Valley Authority (the licensee or TVA) requested amendments to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3. The proposed amendments would change the Technical Specifications to clarify radiation monitoring requirements, remove a redundant limit on spent fuel storage and modify the requirements on the residual heat removal service water (RHRSW) pumps.

2.0 Evaluation

2.1 Radiation Monitoring

Section 6.3.D of the Technical Specifications (T.S.) describes the radiation control procedures for entry into high radiation areas. The present T.S. only require that a person entering a high radiation area "be provided with a radiation monitoring device which continuously indicates the radiation dose while in the area". NRC's resident inspectors have pointed out that this does not provide sufficient measures to preclude potential overexposures. Accordingly, TVA has proposed changes to this section of the T.S. to specify that an individual entering a high radiation area be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area or that stay times in the area be established by qualified Health Physics personnel who have surveyed the area with proper radiation monitoring instrumentation. We conclude that the proposed changes are desirable, will clarify and improve radiation safety and are acceptable.

2.2 Criticality Control on Spent Fuel Storage

The present T.S. include the standard requirement that "the k_{eff} of the spent fuel storage pool shall be less than or equal to 0.95. When the General Electric Company (GE) high density spent fuel storage racks were approved for Browns Ferry (Amendment Nos. 42, 39 and 16 to Facility Licenses

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Nos. DPR-33, DPR-52 and DPR-68, respectively, issued September 21, 1978) an additional limit with respect to criticality of stored spent fuel was added to Section 5.5.B of the T.S.; this limit was set at 15.2 grams of uranium -235 per axial centimeter of fuel assembly "pending an NRC review".

In our review of the proposed use of G.E. high density spent fuel storage racks for Brunswick Unit Nos. 1 and 2, we concluded that BWR fuel having an infinite multiplication factor less than or equal to 1.35 (at its most reactive point in life) in standard core geometry at 20^{o} Centigrade may be safely stored in the G.E. racks such as proposed for Brunswick and installed at Browns Ferry and Monticello. The 1.35 value is required to be calculated for the axial segment of the assembly having the highest reactivity.

The fuel pool criticality analyses performed by GE for Browns Ferry were based on unirradiated BWR fuel with no burnable poison and a fuel loading of 15.2 grams of U-235 per axial centimeter. The maximum calculated fuel storage Keff of 0.87 was derived using a bundle design which has a K_{inf} of 1.35 in the reactor core at 20^{o} for the uncontrolled state. This design value is conservative and is also an upper bound value for the storage racks since fuel with K_{inf}=1.35 will not satisfy the core design requirements for shutdown margin, control worth, etc. The K_{inf} of high enrichment bundles is kept below 1.35 by the use of burnable poisons. Since all bundle types will have maximum K_{inf}'s less than 1.35, the fuel pool Keff limit will be satisfied and a redundant fuel loading limit is not needed.

We conclude that the proposed change is acceptable and will make the Browns Ferry T.S.s consistent in this area with the BWR Standard Technical Specifications (NUREG-0123, Revision 2 dated August 1979).

2.3 <u>RHR Service Water Pump</u>

Section 3.5.C.2 of the T.S. includes a table with footnotes specifying the number of RHRSW and Emergency Equipment Cooling Water (EECW) pumps that must be operable and the periods of time one or more pumps may be out of service. The present T.S. do not distinguish between the requirements for operable RHRSW and EECW pumps for when the plant is operating and when it is in cold shutdown with all fuel removed from the reactor vessel (as is the case during the current Mark I torus modifications). One proposed change is to add another footnote that would reduce from 7 or 6 to 5 or 4 the number of RHRSW pump that must be operable and assigned to service when all fuel is unloaded from a unit reactor vessel. The second proposed change to the footnotes is to delete the requirement that at least one operable RHRSW pump be assigned to each header.

For maintenance and testing purposes, it is sometimes necessary to remove diesel generators and 4-KV shutdown boards from service. Removal of 4-KV shutdown boards or diesel generator A or B will disable RHRSW pumps A1, A2, and C1, C2 respectively. Technical Specification 3.5.C.2, as presently written, requires that at least six RHRSW pumps be available and that at least one pump be aligned to each of the four RHRSW pump headers. Removal of a single shutdown board can be accomplished within the first condition, but the inoperable pump combination conflicts with the requirement for pumps being assigned on each header. If Technical Specification 3.5.C.2 cannot be met, then Technical Specification 3.5.C.6 requires that all three units be placed in cold shutdown within 24 hours.

We have reviewed the applicable configurations regarding RHRSW pump assignment and have concluded that no basis exists which requires that pumps be assigned to all four headers. This is discussed further in the safety evaluation below. The requirement is believed to be erroneously carried over from the same limitation on EECW pump assignment for which there is a basis for header assignment. Also, Technical Specification 3.9.D.4 specifically allows removal of shutdown boards from service for specified time limits to accommodate testing or maintenance. This proposed technical specification will resolve the apparent conflict between the two specifications.

As discussed in the FSAR and technical specification basis 3.5.C, two RHR heat exchangers can more than adequately handle the cooling requirements of one unit in any abnormal or postaccident situation. Additionally, one RHR service water header can adequately deliver the flow supplied by both RHRSW pumps to any two of the three heat exchangers on the header. Therefore, if six RHRSW pumps are operable on three separate headers, all flow requirements for cooling for the three units are met.

If 4-KV shutdown board A is removed from service, RHR pumps 1A and 2A are also removed from service and will not require RHR heat exchanger cooling water from the RHRSW header A; however, the 3A RHR heat exchanger will also lose cooling water. This does not present a problem since only two RHR heat exchangers are required for cooling and three heat exchangers with associated RHRSW remain available. This is the same situation for 4-KV shutdown board B, associated RHR pumps 1C and 2C, and RHR heat exchangers 1C, 2C, and 3C.

Removal of a single 4-KV shutdown board is presently allowed in Technical Specification 3.9.B.4.

Based on the above, we have concluded that the requirement to always have one operable RHRSW pump assigned to each header is not warrented and should be deleted from T.S. Section 3.5.C.2. We have also concluded that when all fuel is removed from reactor vessel-which can only be accomplished after the unit has been in cold shutdown for several weeks- that 4 or 5 RHRSW pumps is more than adequate for any projected cooling loads on these systems.

2.4 Cumulative Downtime

The definitions in Section 1.0 of the Browns Ferry T.S. presently include a definition on cumulative downtime that states that: "The cumulative downtime for those safety components and systems whose downtime is limited to 7 consecutive days prior to requiring reactor shutdown shall be limited to any 7 days in a consecutive 30 day period". No similar definition is included in the BWR Standard Technical Specifications. TVA has proposed that this definition be removed from the Browns Ferry T.S.

The bases for not including such a definition in the BWR Standard Technical Specifications and for deleting it from the older T.S. was discussed in a memorandum dated August 16, 1977 from Karl R. Goller, Assistant Director for Operating Reactors, to J. H. Sniezek, Assistant Director, Division of Reactor Operations Inspection. The memorandum stated that:

"Action statements were developed to accommodate those instances when equipment, components or other specific conditions of the specifications could not be met because of whatever reason. We recognized then, at well as now, that the potential existed for licensees to take advantage of these provisions in order to perform activities within the three categories you describe. At that time, we considered the following in order to restrict such activities:

a. Limiting the length of time that specified components or systems may remain inoperable before further action would be required, and

b. Limiting the number of times and/or the total cumulative length of time during a specified period of time that specified components or systems may be inoperable.

However, in view of the complex and extensive record keeping problems and the lack of an adequate data base from which to infer acceptable limiting outage periods, we did not consider the benefits to be gained justifiable when balanced against the increased effort required by licensees and I&E inspection personnel. Additionally, we believed that we would be able to remain cognizant of possible abuse of outage times through review of LER's, supplemented where necessary, by notification action of the I&E inspector assigned to each facility."

In view of the policy position enumerated above, we conclude that the proposed change is acceptable.

3.0 Environmental Considerations

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

4.0 Conclusion

We have concluded based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards 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: August 6, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 74 to Facility Operating License No. DPR-33, Amendment No. 71 to Facility Operating License No. DPR-52, and Amendment No. 46 to Facility Operating License DPR-68 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

These amendments change the Technical Specifications to clarify radiation monitoring requirements, remove a redundant limit on spent fuel storage and modify the requirements on the residual heat removal service water pumps.

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 April 28, 1981, (2) Amendment No. 74 to License No. DPR-33, Amendment No. 71 to License No DPR-52, and Amendment No. 46 to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room , 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 6th day of August, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

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