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

MAY 11 1979

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 Amendments Nos. 51, 45 and 23 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 December 28, 1977, as supplemented by your letter of December 13, 1978. The Amendments add a condition to the license for each facility authorizing you to perform the modifications, as described in your submittals, to change the power supply for certain low pressure coolant injection (LPCI) valves for Units Nos. 1, 2 and 3 and to eliminate the loop selection logic for Unit No. 3. In your letter of December 13, 1978, you propose to install the M-G sets by the end of the second refueling outage of Unit No. 3 and the third refueling outages of Units Nos. 1 and 2. As committed to in your letter of June 15, 1977, proposed Technical Specification changes associated with the modifications approved herein are to be submitted with the reload amendment request for each Unit. As agreed to by your staff, you will also submit a revision to the Browns Ferry Nuclear Plant Final Safety Analysis Report (FSAR) when the modifications are completed to document the changes in plant design.

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

Sincerely,

Original signed By

Contract CLP

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

7906250088 P

*SEE PREVIOUS YELLOW FOR CONCURRENCES

Enclosures:
1. Amendment No. 51 to DPR-33
2. Amendment No. 45 to DPR-52
3. Amendment No. 23 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures:
see next page

OFFICE	ORB#3	ORB#3	AD/E&P/DOR	OELD	ORB#3
SURNAME	*SSheppard	*RClark:ar	*BGrimes	*MCutchin	TIppolito
DATE	4/19/79	4/19/79	4/19/79	5/4/79	5/11/79

NRC FORM 318 (9-76) NRCM 0240 U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

Docket Nos. 50-259
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Mr. Hugh G. Parris
Manager of Power
Tennessee Valley Authority
500 A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

The Commission has issued the enclosed Amendments Nos. , and 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 December 28, 1977, as supplemented by your letter of December 13, 1978. The Amendments add a condition to the license for each facility authorizing you to perform the modifications, as described in your submittals, to change the power supply for certain low pressure coolant injection (LPCI) valves for Units Nos. 1, 2 and 3 and to eliminate the need for the swing bus feature for Unit No. 3.

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

Sincerely,

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

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

cc w/enclosures:
see next page

OFFICE	ORB#3	ORB#3	AD/E&P/DOR	OELD	ORB#3	
SURNAME	SSheppard	RClark:acr	BGrimes	see memo of 5/4 Cutchin to Clark	TIppolito	
DATE	4/19/79	4/19/79	4/19/79	5/4/79	4/19/79	

NRC FORM 318 (9-76) NRCM 0240 U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

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Atlanta, Georgia 30308

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Decatur, Alabama 35602

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. 51
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 December 28, 1977, 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.

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2. Accordingly, paragraph 2.C of Facility License No. DPR-33 is hereby amended by adding subparagraph (9) as follows:

   (9) The facility may be modified as described in 'Browns Ferry Nuclear Plant Units 1 and 2 Emergency Core Cooling Systems Low Pressure Coolant Injection Modifications For Performance Improvement (October 1977)' submitted by letter dated December 28, 1977 and supplemented by letter dated December 13, 1978.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

Date of Issuance: May 11, 1979

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. 45
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 December 28, 1977, 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, paragraph 2.C of Facility License No. DPR-52 is hereby amended by adding subparagraph (9) as follows:

   (9) The facility may be modified as described in 'Browns Ferry Nuclear Plant Units 1 and 2 Emergency Core Cooling Systems Low Pressure Coolant Injection Modifications for Performance Improvement (October 1977)' submitted by letter dated December 28, 1977 and supplemented by letter dated December 13, 1978.

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

Date of Issuance: May 11, 1979

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. 23
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 December 28, 1977, 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, paragraph 2.E of Facility License No. DPR-68 is hereby amended by adding subparagraph (6) as follows:

   (6) The facility may be modified as described in 'Browns Ferry Nuclear Plant Unit 3 Emergency Core Cooling Systems Low Pressure Coolant Injection Modifications for Performance Improvement (October 1977)' and as described in TVA's letter of December 28, 1977 transmitting the aforementioned report and in TVA's supplemental letter of December 13, 1978.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Date of Issuance: May 11, 1979

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 23 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

By letter dated December 28, 1977 and supplemented by letter dated December 13, 1978, the Tennessee Valley Authority (the licensee or TVA) submitted proposed design modifications for the Low Pressure Coolant Injection (LPCI) systems of the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3 and a detailed safety analysis of the modification.

For Units Nos. 1 and 2 the licensee's proposed modification consists of changing the power supply to the motor operators of certain LPCI system valves. The change involves the use of Class 1E motor-generator sets as isolation devices between the swing bus of the 480 V reactor MOV boards that supply power to the valve operators and the divisional 480 V shutdown boards.

For Unit No. 3, the licensee's proposed modification consists of the following:

a. Elimination of the Low Pressure Coolant Injection (LPCI) system's recirculation loop selection logic, revision of the logic and closure of the Residual Heat Removal (RHR) cross-tie valve and a recirculation equalizer valve; and

b. Changing the power supply to the recirculation pump discharge valves, LPCI injection valves, RHR pump minimum flow bypass valves, and RHR test isolation valves. The change also modifies independent valve a.c. power supplies to eliminate concerns on paralleling of divisional a.c. power supplies, and modifies d.c. power supplies to 4kV shutdown board control.

Item a, above, of the proposed modification is similar to those modifications previously approved by the staff and implemented at the Browns Ferry Nuclear Plant, Units 1 and 2 (BFNP-1 & 2) (Amendments Nos. 27 and 24 for Units Nos. 1 and 2, respectively, dated August 20, 1976). Item b, above, is in line with the modifications proposed herein for Units Nos. 1 and 2.

2.0 Discussion

1. Units Nos. 1 and 2

The proposed design modification to the LPCI systems of the Browns Ferry Nuclear Plant, Units 1 and 2, (BFNP 1 & 2) was submitted in accordance with a licensee agreement made when Amendment No. 27 to Facility License No. DPR-33 and Amendment No. 24 to Facility License No. DPR-52 were issued. The amendments, in part, dealt with the elimination of the LPCI system's recirculation loop selection logic and closure of the RHR cross-tie valve. The proposed modification is designed to assure that the 480 V ac reactor MOV boards, with the associated auto-transfer feature, will be isolated from the redundant divisional power supplies.

In the existing LPCI system, redundant LPCI injection valves, recirculation pump discharge valves, RHR pump minimum flow valves and RHR test isolation valves are connected to separate 480 V reactor MOV boards and are supplied power from redundant power supplies, i.e., Diesel Generator A and Diesel Generator C for Unit 1 and Diesel Generator B and Diesel Generator D for Unit No. 2. The design is such that upon loss of normal power supply to a reactor MOV board, an automatic transfer connects the MOV board to its redundant power supply. This automatic transfer scheme compromises the independence between redundant power supplies. Although analysis had shown that no single failure in the interlocks that effect automatic transfer between divisions would adversely affect redundant divisions of power supply, the licensee had committed to modify the power distribution system to eliminate the concerns regarding paralleling of redundant power supplies.

The proposed modification changes the power supply to the reactor MOV boards that feed the motor operators of the LPCI injection valves, the recirculation pump discharge valves, and the RHR pump minimum flow bypass valves. The change involves the use of Class 1E motor-generator (M-G) sets as isolation devices between the auto-transfer feature of the 480 V reactor MOV boards and the divisional 480 V shutdown boards. The auto-transfer feature will be eliminated from all 480 V reactor MOV boards not protected by M-G sets.

As part of the modified LPCI system power supply the licensee will provide redundant M-G sets between the divisional 480 V shutdown boards and each 480 V reactor MOV board. The auto-transfer feature of each 480 V reactor MOV board will be retained. For example, for BFNP-1, 480 V Reactor MOV Board 1D will be normally supplied by motor-generator set MG-1DN, connected to 480 V Shutdown Board 1A, with alternate supply to be available from M-G set MG-1DA connected to 480 V Reactor MOV Board 1B. Similarly, 480 V Reactor MOV Board 1E will have MG-1EN as the normal power supply source connected to 480 V Shutdown Board 1B, and MG-1EA as the alternate source connected to 480 V Shutdown Board 1A. The arrangement of the 480 V reactor MOV boards and the M-G sets for BFNP-2 will be similar.

The M-G sets are designated as Class 1E equipment and will be designed to seismic Category I standards. Each M-G set will be sized to accept the load requirements of the valve operators at any time during the initiating event. The sets will be designed to operate within design specifications when supplied by the diesel generators. Each M-G set will act as an isolating device between the 480 V shutdown board and the reactor MOV board. Overload protection of the motor and generator will be separately provided for each set. Control for the M-G sets will be at the 480 V shutdown boards and loss of each M-G set output voltage will be annunciated in the main control room. Although only one M-G set will normally supply power to each 480 V reactor MOV board, both M-G sets will run at all times to assure readiness of the alternate M-G set to accept load. The auto-transfer feature of each reactor MOV board will be retained to assure power to the valve operators. The auto-transfer scheme has already been analysed to ensure that a single failure in the circuit will not affect redundant divisions of power. The insertion of isolation M-G sets between the Reactor MOV boards and the shutdown boards provides added assurance of independence between redundant divisions of power supply. For the reactor MOV boards which do not have the auto-transfer feature interlocks between the divisional supply breakers, use of redundant series feeder breakers at the 480 V shutdown boards assure that a single failure will not compromise divisional power supplies.

The RHR test isolation valve operators were not included in the previous modification of the LPCI system. By the proposed modifications, the RHR test isolation valves will be provided with redundant power supplies and closing control logics.

2. Unit No. 3

The existing Low Pressure Coolant Injection (LPCI) mode of the RHR system at the Browns Ferry Nuclear Plant, Unit 3 (BFNP-3) is the standard BWR-4 configuration, using four pumps and a loop selection logic. The LPCI injection valves are closed and the RHR cross-tie valve is open during normal operations. On receipt of an accident initiation signal following a recirculation line break in one loop, the loop selection valve in that loop is signaled to open, the recirculation pump discharge valve in that loop is signaled to close, and LPCI flow from all four pumps is directed to the unbroken loop.

The proposed modification to the LPCI system of BFNP-3 was submitted by the licensee to improve the Emergency Core Cooling System (ECCS). The proposed modification involves the following changes:

a) The recirculation loop selection logic will be eliminated, and the accident initiation signals will be rewired to direct both LPCI injection valves to open upon detection of accident conditions.

b) Both recirculation loop discharge valves will be signaled to close when the reactor pressure decreases to an appropriate setting following detection of accident conditions.

c) The cross-tie valve between the two RHR system headers and a recirculation equalizer valve will be kept closed and an annunciator added to indicate any not-fully-closed condition. The motive power to the valves will also be disconnected.

d) The auto-transfer feature of the 480 V reactor MOV boards that supply motive power to the LPCI valves, will be isolated from redundant divisional power sources by motor-generator (M-G) sets.

e) A qualified battery that supplies dc control power to a 4kV shutdown board will be added, and the dc control power source to another 4kV shutdown board will be changed.

3.0 Evaluation

1. Units Nos. 1 and 2

The proposed modification has been designed to seismic Category I standards. The modified system design has been reviewed against the standards and guides which were applicable to the original design to assure that the modified system design, equipment and installation meet or exceed the qualification of the unmodified system, including seismic qualification. The licensee has committed to apply quality assurance and control to this modification in accordance with the requirements of 10 CFR 50, Appendix B.

The licensee has also submitted certain analyses which we have reviewed that were performed in accordance with the requirements of 10 CFR 50, Appendix K to consider the emergency core cooling performance with operation of the modified power systems. Based on the analysis for a recirculation loop pipeline break, the limiting single failure (which resulted in the highest peak cladding temperature [PCT]), was seen to be the suction line break with the failure of the RHR injection valve in the unbroken loop. For this condition, the resulting peak clad temperature (PCT) was below the allowable PCT limit. (Table 1 shows the ECCS pump configuration for various postulated single failures).

We have also reviewed the licensee's analysis of the single failure which might influence the long-term suppression pool cooling mode of the modified RHR system. For the worst case single failure, the suppression pool temperature was found to be within allowable limits. The analysis and evaluation were done to assure that the changes do not introduce adverse effects to the overall plant. This investigation considered the effects on the capability of major affected equipment (e.g., Diesel Generators, dc batteries, RHR pumps, and RHR system valves) and on the operating modes of the affected equipment (Diesel Generator Control, RHR Logic Panels and DC Control Power).

Based on our review of the proposed modifications to Units Nos. 1 and 2, we find that:

a. The proposed modification to the Low Pressure Coolant Injection (LPCI) system will assure that the relevant sections of the onsite power system have sufficient independence between redundant power sources, and thus meets the requirements of General Design Criterion 17. This has been accomplished by the provision of redundant Class 1E motor-generator sets to act as isolation power supply devices between the 480 V shutdown boards and the 480 V reactor MOV boards.

b. The additional analyses performed in accordance with the requirements of 10 CFR 50, Appendix K, to consider ECCS performance with the modified power have confirmed that for the design basis LOCA and assuming the worst single failure, the peak cladding temperature is below the allowable limit.

c. The proposed changes do not introduce adverse effects to the overall plant.

Accordingly, we conclude that the proposed design modifications for Units Nos. 1 and 2 are acceptable.

2. Unit No. 3

The proposed modifications to LPCI system for Unit No. 3 were described in the Discussion section. Our evaluation of each proposed modification is summarized below.

a. Elimination of Loop Selection Logic

Elimination of the loop selection logic and rewiring of the logic circuitry will direct both LPCI injection valves to open, irrespective of the location of the break in the recirculation loop. The start logic of the RHR pumps will be changed by the addition of redundant start commands to all pumps and the operating modes will be changed such that two pumps discharge to each injection header. The wiring changes for elimination of the loop selection logic and the rewiring required are to the same standards applied to the original design. All standards for engineered safety feature control equipment will also be maintained. Additional relays and wiring will be added to assure single failure capability. Orifices for additional flow resistance will be installed in the RHR pump discharge lines to limit the maximum pump flow when the RHR pumps discharge to the broken loop. The information obtained from tests that were conducted on similar pumps of BFNP-1 and BFNP-2 will be used to determine the additional resistance to be added on the discharge side of each BFNP-3 pump to ensure that the pumps' Net Positive Suction Head (NPSH) requirements are satisfied.

b. Recirculation Loop Valves

Recirculation pump discharge valve closure requires both a LOCA initiation signal and a decrease in reactor pressure to the permissive setting. With valve closure initiation delayed until reactor pressure has decayed to less than 225 psig, the differential pressure across the closed valve will always be less than 200 psia, (i.e., within the capability of the valve). The sensor and permissive circuitry will be designed to satisfy all requirements for engineered safety feature control systems.

c. RHR Cross - Tie Valve and Recirculation Loop Equalizer Valve

The RHR system cross-tie valve and a recirculation loop equalizer valve will be kept closed and motive power to the valve operators removed to prevent any inadvertent opening of these valves that could negate RHR system injection when needed.

d. Motor-Generator Sets

Qualified motor-generator (M-G) sets will be used as isolation devices on the 480 V reactor MOV boards with auto-transfer feature. These MOV boards supply motive power to those valves necessary for automatic operation of RHR injection (recirculation pump discharge valves, LPCI injection valves, RHR pump minimum flow bypass valves and RHR test isolation valves) and will interface with the divisionalized 480 V shutdown boards through the M-G sets. Each MOV board will have two sets, and although only one M-G set will normally supply power to the MOV board, both M-G sets will run at all times to assure readiness of the alternate M-G set to accept load if required. Each M-G set will be sized to accept valve loads at any time during the initiating event. The M-G sets will be designed to operate within design specifications when supplied by the diesel generators. Overload protection of the motor and generator will be separately provided for each set

Control for the M-G sets will be at the 480 V shutdown boards and loss of each M-G set output voltage will be annunciated in the main control room. The auto-transfer feature of the reactor MOV boards will be retained to assure power to the valve operators. The auto-transfer scheme has already been analysed to assure that a single failure in the circuit will not affect redundant divisions of power. For those reactor MOV boards which do not have the auto-transfer scheme, interlocks between the divisional supply breakers and use of redundant feeder breakers at the 480 V shutdown boards assure that a single failure will not compromise divisional power supplies.

e. <u>250 V D.C. Battery</u>

A separate qualified 250 V dc battery will be added to provide control power to 4kV Shutdown Board 3EB. The 250 V dc control power source to 4kV Shutdown Board 3ED will be changed to a different station battery. These changes will assure that control power for all four 4kV shutdown board breakers are fed from different sources to meet the single failure criterion. The design of the proposed battery will be to original standards for similar batteries on BFNP-1 and BFNP-2. Ventilation and fire protection systems will be provided for the proposed battery as required.

The proposed modifications for Unit No. 3 have been designed to seismic Category I standards. We have reviewed the modified system design against the standards and guides which were applicable to the original design and have assured that the modified system design, equipment and installation meet or exceed the qualification of the unmodified system, including seismic qualification. The licensee has committed to apply quality assurance and control to these modifications in accordance with the requirements of 10 CFR 50, Appendix B.

The licensee has also submitted certain analyses which we have reviewed that were performed in accordance with the requirements of 10 CFR 50, Appendix K to consider the emergency core cooling performance with operation of the modified power system. Based on the analysis for a recirculation loop pipeline break, the limiting single failure (which resulted in the highest peak cladding temperature [PCT]), was seen to be the suction line break with the postulated failure of the RHR injection valve in the unbroken loop. For this condition, the resulting peak clad temperature (PCT) was below the allowable PCT limit. (Table 2 shows the ECCS pump configuration for various postulated single failures).

The single failure which might influence the long-term suppression pool cooling mode of the modified RHR system has also been analyzed. For the worst case single failure, the suppression pool temperature was found to be within allowable limits. The analysis and evaluation were done to assure that the changes do not introduce adverse effects to the overall plant. This investigation considered the effects on the capability of major affected equipment (e.g., Diesel Generators, dc batteries, RHR pumps, and RHR system valves) and on the operating modes of the affected equipment (Diesel Generator Control, RHR Logic Panels and DC Control Power).

Based on our review of the proposed modifications to the LPCI system for Unit No. 3 we find that:

a. The elimination of the LPCI system loop selection logic and rewiring of the ECCS initiation signals will assure that the modified circuits meet the single failure criteria. This has been accomplished by the use of redundant and separate relays and wiring in the RHR logic panels.

b. For the recirculation loop discharge valve, the valve closure initiation is delayed until reactor pressure has decayed to less than 225 psig. This ensures that the differential pressure across the closed valve will always be less than 200 psid. Further, the sensor and permissive circuitries are designed to satisfy all requirements for engineered safety feature control systems.

c. The proposed revisions assure that the RHR cross-tie valve and a recirculation loop equalizer valve will remain closed during normal plant operations and accident conditions. This has been done by disconnecting valve motive power with the valves closed and by providing an annunciator to indicate when a valve is not fully closed.

d. The provision of redundant Class IE motor-generator sets which function as isolation devices between the 480 V shutdown boards and the new 480 V reactor MOV boards will assure that the relevant sections of the onsite power systems have sufficient independence between redundant power sources.

e. The addition of a qualified battery to supply control power to a 4kV shutdown board and the change of dc control power source of another 4kV shutdown board will assure that the modified LPCI system will meet the postulated single failure of a dc power source.

f. The proposed changes do not introduce adverse effects to the overall plant.

g. The modifications brings the BFNP-3 Emergency Core Cooling System (ECCS) in line with the ECCS of BFNP-1 and BFNP-2.

Accordingly, we conclude that the proposed design modifications for Unit No. 3 are 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 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.

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: May 11, 1979

TABLE 1

ECCS PUMP CONFIGURATION

Suction Side Break	Pumps Available**
No Failures	4 Core Spray, 2 RHR in each Loop
Opposite Unit Spurious Accident Signal	2 Core Spray, 1 RHR in each Loop
RHR Injection Valve Failure*	4 Core Spray, 2 RHR in one Loop
RHR Minimum Valve Failure*	4 Core Spray, 2 RHR in one Loop
Recirculation Discharge Valve Failure-Break Side*	4 Core Spray, 2 RHR in one Loop
480 V Reactor MOV Board*	4 Core Spray, 2 RHR in one Loop
Diesel Failure	2 Core Spray, 2 RHR in one Loop, 1 RHR in other Loop
Battery Failure	2 Core Spray, 2 RHR in one Loop, 1 RHR in other Loop

Discharge Side Break	Pumps Available**
No Failures	4 Core Spray, 2 RHR in one Loop
RHR Injection Valve Failure*	4 Core Spray
RHR Minimum Flow Valve Failure*	4 Core Spray
Diesel Failure	2 Core Spray, 1 RHR
Battery Failure	2 Core Spray, 1 RHR
Opposite Unit Spurious Accident Signal	2 Core Spray, 1 RHR
480 V Reactor MOV Board*	4 Core Spray

*Limiting Single Failure
**In Unbroken Loop

TABLE 2

ECCS PUMP CONFIGURATION IN MODIFIED SYSTEM

Suction Side Break	Pumps Available**
No Failures	4 Core Spray, 2 LPCI in each Loop
LPCI Injection Valve Failure*	4 Core Spray, 2 LPCI in one Loop
LPCI Minimum Valve Failure*	4 Core Spray, 2 LPCI in one Loop
Recirculation Discharge Valve Failure-Break Side*	4 Core Spray, 2 LPCI in one Loop
480 V Reactor MOV Board Failure*	4 Core Spray, 2 LPCI in one Loop
Diesel Failure	2 Core Spray, 2 LPCI in one Loop, 1 LPCI in other Loop
Battery Failure	2 Core Spray, 2 LPCI in one Loop, 1 LPCI in other Loop

Discharge Side Break	Pumps Available**
No Failures	4 Core Spray, 2 LPCI in one Loop
LPCI Injection Valve Failure*	4 Core Spray
LPCI Minimum Flow Valve Failure*	4 Core Spray
480 V Reactor MOV Board Failure*	4 Core Spray
Diesel Failure	2 Core Spray, 1 LPCI
Battery Failure	2 Core Spray, 1 LPCI

*Limiting Single Failure
**In Unbroken Loop

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. 51 to Facility Operating License No. DPR-33, Amendment No. 45 to Facility Operating License No. DPR-52 and Amendment No. 23 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), 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.

The amendments add a condition to the license for each facility authorizing TVA to improve the performance of the emergency core cooling systems by changing the power supply to the low pressure coolant injection (LPCI) system in each Unit and to modify the Unit No. 3 loop selection logic circuitry.

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 Section 51.5(d)(4) an 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 December 28, 1977, (2) Amendment No. 51 to License No. DPR-33, Amendment No. 45 to License No. DPR-52, and Amendment No. 23 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 May 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

T. Ippolito

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