Docket

MAR 11 1981

Distribution
Docket
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SNorris
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TBarnhardt (12)
BScharf (10)
JWetmore
ACRS (16)
OPA (CMiles)
RDiggs
HDenton
JHeltemes, AEOD
NSIC
TERA

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 Amendments Nos. 70, 66 and 42 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 consist of changes to the Technical Specifications in response to your request of March 22, 1978 (BFNP TS 107), as supplemented by your letter of October 10, 1978.

The amendments modify the Technical Specifications to revise note 7 in Table 3.2.B regarding the acceptance criteria in the startup test instructions for the High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems. With the concurrence of your staff, we have made minor changes in the wording for note 7 from that which you submitted with your letter of March 22, 1978.

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 #2
Division of Licensing

Enclosures:
1. Amendment No. 70 to DPR-33
2. Amendment No. 66 to DPR-52
3. Amendment No. 42 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures: See page 2

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

March 11, 1981

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 Amendments Nos. 70, 66 and 42 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 consist of changes to the Technical Specifications in response to your request of March 22, 1978 (BFNP TS 107), as supplemented by your letter of October 10, 1978.

The amendments modify the Technical Specifications to revise note 7 in Table 3.2.B regarding the acceptance criteria in the startup test instructions for the High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems. With the concurrence of your staff, we have made minor changes in the wording for note 7 from that which you submitted with your letter of March 22, 1978.

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

Sincerely,

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

Enclosures:
1. Amendment No. 70 to DPR-33
2. Amendment No. 66 to DPR-52
3. Amendment No. 42 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures: See page 2

cc:

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Decatur, Alabama 35602

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Knoxville, Tennessee 37902

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Tennessee Valley Authority
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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. 70
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 March 22, 1978, as supplemented by letter dated October 10, 1978, 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. 70, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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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: March 11, 1981.

ATTACHMENT TO LICENSE AMENDMENT NO. 70

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

71/72

2. The underlined page is the page being changed; the marginal line on this page indicates the revised area. The overleaf page is provided for convenience.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by section 3.5 to be operable, there shall be two operable trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not operable in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required, indicators are considered redundant.

2. In only one trip system.

3. Not considered in a trip system.

4. Requires one channel from each physical location (there are 4 locations) in the steam line space.

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 sec later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec with similar pumps starting after about 14 sec and 21 sec, at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150% above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150% above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.1, which ensures water in the discharge piping and up to the head tank.

NOTES FOR TABLE 3.2.B (Continued)

10. Only one trip system for each cooler fan.

11. In only two of the four 4160 V shutdown boards. See note 13.

12. In only one of the four 4160 V shutdown boards. See note 13.

13. An emergency 4160 V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level ($\geq$ 378" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one operable trip system. Therefore one trip system may be taken out of service for functional testing and calibration for a period not to exceed 8 hours.

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. 66
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 March 22, 1978, as supplemented by letter dated October 10, 1978, 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. 66 , 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: March 11, 1981.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by section 3.5 to be operable, there shall be two operable trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not operable in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required, indicators are considered redundant.

2. In only one trip system.

3. Not considered in a trip system.

4. Requires one channel from each physical location (there are 4 locations) in the steam line space.

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 sec later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec with similar pumps starting after about 14 sec and 21 sec, at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150% above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150% above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.1, which ensures water in the discharge piping and up to the head tank.

NOTES FOR TABLE 3.2. (Continued)

10. Only one trip system for each cooler fan.

11. In only two of the four 4160 V shutdown boards. See note 13.

12. In only one of the four 4160 V shutdown boards. See note 13.

13. An emergency 4160 V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level ($\geq$ 378" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one operable trip system. Therefore one trip system may be taken out of service for functional testing and calibration for a period not to exceed 8 hours.

ATTACHMENT TO LICENSE AMENDMENT NO. 66

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

71/72

2. The underlined page is the page being changed; the marginal line on this page indicates the revised area. The overleaf page is provided for convenience.

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. 42
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 March 22, 1978, as supplemented by letter dated October 10, 1978, 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. 42, 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: March 11, 1981.

ATTACHMENT TO LICENSE AMENDMENT NO. 42

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

74

2. The marginal line on the above page indicates the area being changed.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by section 3.5 to be operable, there shall be two operable trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not operable in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required, indicators are considered redundant.

2. In only one trip system.

3. Not considered in a trip system.

4. Requires one channel from each physical location (there are 4 locations) in the steam line space.

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 sec later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec with similar pumps starting after about 14 sec and 21 sec, at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150% above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150% above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.1, which ensures water in the discharge piping and up to the head tank.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 42 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 March 22, 1978, and supplemented by letter dated October 10, 1978, 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 change the acceptance criteria for determining the adequacy of the setpoints for a postulated break in the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) steam lines.

2.0 Discussion

The HPCI and RCIC systems as described in the Browns Ferry Nuclear Plant Final Safety Analysis Report (FSAR) are steam turbine powered systems which are capable of pumping water into the reactor at full system pressure. They are activated by reactor low water level signals. The steam supply for these systems comes from the main steam header upstream of the main steam line isolation valves. It is necessary to isolate the RCIC and HPCI steam supply lines in the event of a rupture of these lines outside the primary containment. In addition to equipment space temperature instrumentation to detect a break in either the RCIC or HPCI steam lines, another means for detecting HPCI and RCIC steam line breaks outside containment is steam mass flow instrumentation upstream of the system isolation valves. This instrumentation provides a closure signal to the steam line isolation valves when steam flow rises significantly above steady conditions.

The current Technical Specifications for the Browns Ferry Nuclear Plant (BFNP) lists in Table 3.2.B (page 62 for Units 1 and 2, page 64 for Unit 3) the "Instrumentation that Initiates or Controls the Core and Containment Cooling Systems". Note 7 to this table contains a statement that the "RCICS setting of 450 inches of H_2O corresponds to 300% of rated steam flow at 1140 psia and 210% at 165 psia" and that the "HPCIS setting of 90

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psia corresponds to 225% of rated flow at 1140 psia and 160% at 165 psia". The proposed change would eliminate reference to exact percentages of steam flow but state that the current settings of 450" of water and 90 psia for the RCICS and HPCIS, respectively, provide a substantial margin above steady state (steam) flow to assure that spurious isolation does not occur while also ensuring the initiation of isolation following a postulated steam line break. The licensee is proposing that the setpoints of the steam flow instrumentation remain the same as at present, so there would be no change in the facilities themselves. The proposed change is in the acceptance criteria in the startup test instructions; these criteria have been reevaluated as a result of information developed during the startup tests of the Browns Ferry units. These tests showed that the steady state mass steam flow is dependent upon the operating conditions.

The HPCI steam line flow is measured at an orifice located upstream of the isolation valves. The instrumentation is calibrated to initiate isolation when a differential pressure of 90 psi is measured across the orifice. Startup tests under conditions of highest steady state steam flow showed that with the 90 psi setpoint, the trip would have occurred at a minimum value of 166 percent of the steady state flow rate (rather than occurring precisely at a certain percentage of rated steam flow as the present Note 7 to Table 3.2.B implies).

RCIC steam line flow is measured at an elbow located upstream of the isolation valves. The instrumentation is calibrated to initiate when a differential pressure of 450 inches of water is measured across the elbow. Startup tests under conditions of highest steady state steam flow showed that with the 450 inches of water setpoint, the isolation trip would have occurred at a minimum value of 156 percent of the steady state flow rate rather than precisely at a certain percentage of rated steam flow as Note 7 implies. The licensee has stated that calculations show that the present setpoints will be exceeded if a RCIC or HPCI steam line break were to occur thus assuring isolation. Raising the setpoints could decrease the margin available for detection of a steam line break. The licensee believes leaving them as they are would continue to provide substantial margins above maximum steady state flows which would very likely never be exceeded and the smaller margins indicated by some of the test results do not jeopardize the normal performance of the systems.

A major difficulty is that the steady state mass steam flow to the turbine of either system under some conditions of reactor pressure, pump discharge pressure, pump suction pressure, etc., was found in the startup tests to be as much as twice the flow as under other conditions. A further complication is that the actual setpoint corresponds to a fixed differential pressure measured across an orifice or elbow. Therefore, the criteria that steam line isolation is to be initiated at a given fixed percentage of steady state mass steam flow is impractical. As noted above, in the startup tests under conditions of highest steady state mass steam flows, the trip point

would have occurred at minimum values of 156 and 166 percent of the steady state mass steam flow values for the RCIC and HPCI systems respectively. Also, these tests indicated that rarely would the systems have such low steady state steam flow values that the trip point would have been as high as 300 and 225 percent of the steady state values respectively.

3.0 Evaluation

The only proposed changes to the Technical Specifications involve changes in procedures. The setpoints of the instruments remain the same, so no change is to be made to the facility itself. The tests themselves were done as described in the FSAR. The acceptance criteria have been reevaluated in light of the steady state mass steam flow being dependent upon the operating conditions.

The acceptance criteria of an isolation trip at 300 percent maximum required steady state mass flow for the RCIC and of 225 percent mass flow for the HPCI are the points in question. The function of the trip is to isolate the systems from the reactor in the event of a pipe break in the steam supply line between the primary containment boundary and the steam driven turbine that operate the RCIC and HPCI pumps.

Automatic isolation of the HPCI system occurs on any of the following signals:

1. Reactor pressure below 100 psig (automatic reset).

2. High HPCI steam line flow (300 percent or greater).

3. High temperature in the HPCI steam line spaces.

4. High exhaust rupture diaphram pressure.

High temperature in the vicinity of the HPCI System equipment or a high steam flow to the turbine driven HPCI pump could indicate a break in the HPCI turbine steam supply line. Detection of a high temperature or high steam flow automatically closes certain group A isolation valves to prevent excessive loss of reactor coolant and to prevent the release of radioactive steam outside containment.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in four trip systems with four temperature switches in each trip system. The four temperature switches in each trip system are arranged in one-out-of-two taken twice logic.

As noted in the Discussion, high flow in the HPCI turbine steam line is sensed by two differential pressure switches which monitor the differential pressure across a mechanical flow element installed in the HPCI turbine steam pipeline. The tripping of either switch initiates isolation of the HPCI turbine steam line.

Automatic isolation of the RCIC system occurs on any of the following signals:

1. Low Reactor Pressure

2. High RCIC Steam Line Flow

3. High Temperature in the RCIC Steam Line Space

High temperature in the RCIC Steam Line areas or high steam flow to the RCIC turbine could be indicative of a break in the steam line to the RCIC turbine. Detection of a high temperature or steam flow automatically closes certain group A valves for the same reasons stated above for the HPCI system.

Measurement of temperatures in the RCIC steam line spaces is similar to the design for the HPCI steam line. Temperatures are sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in four trip systems with four temperature switches in each trip system. The four temperature switches in each trip system are arranged in one-out-of-two taken twice logic.

High flow in the RCIC turbine steam line is sensed by two differential pressure switches which monitor the differential pressure across an elbow installed in the RCIC turbine steam supply pipelines. The tripping of either trip channel initiates isolation of the RCIC turbine steam line.

As is evident from the above discussion, there are two independent means of detecting potential breaks in the steam supply lines to the HPCI and RCIC turbines. Both detection systems (temperature and steam flow) can isolate the steam supply lines.

In view of the startup test results, the staff's review concentrated on two potential concerns related to the proposed change to the Technical Specification - (1) is there still reasonable assurance that a break or significant leak in the steam lines to the HPCI and RCIC turbine outside containment will result in isolation of these steam lines and (2) since the apparent margin available before inadvertent isolation occurs is decreased, is there likely to be unnecessary challenges to these safety systems and increased likelihood that these two safety systems might be inadvertently isolated and not available to serve their intended function.

With respect to the first concern, the test data indicates there is increased assurance that the steam lines will isolate if there is a significant break in either line. The primary means of detecting a steam line break - the temperature monitoring system - is considered to be a reliable system; our rereview of the setpoints on these systems concluded that these systems will provide rapid detection of any significant steam leak.

To protect against startup steam flow transients causing inadvertent isolation, two protection systems are provided. First, as noted in the Browns Ferry FSAR, the high differential pressure signal for HPCI goes through a time delay of three seconds before energizing the contacts which cause the HPCI steamline isolation valves to close. Secondly, during the first twelve seconds of operation, the HPCI throttle control valve is gradually opened at a controlled ramp increase to the anticipated normal operating position. At this point, automatic flow control takes over. Data provided by the licensee in their letter of October 10, 1978 verify that these controls function to prevent inadvertent isolation during the initial HPCI startup transient although transient steam flow exceeds the isolation setpoint.

We also determined that to date, there has not been an inadvertent closure of the HPCI or RCIC steam lines due to the present setpoints or malfunctioning of the detection systems.

Based on the controls available and the test data provided, it is the judgement of the staff that the controls are sufficient to prevent inadvertent HPCI isolation. It is the staff judgement that the decrease in the margin available before inadvertent isolation as provided in the proposed Technical Specifications results in a very small increase in the probability of inadvertent isolation. In our judgement, the operability of the system is not significantly altered. Therefore, we believe that the current setpoint for HPCI should be maintained and the startup test criteria changed in order to preserve the current safety level.

As stated above, it is the staff judgement, based on the need to assure isolation of a HPCI and a RCIC system steam line break and to prevent a spurious isolation of HPCI, that the current setpoints are adequate and should be maintained. However, the results of the testing done at Browns Ferry indicate that to maintain the current acceptable setpoints, the criteria stated in Note 7 for Table 3.2.B of the Browns Ferry Technical Specifications should be changed. The test information provided in the licensee's submittal of March 22, 1978 and in their responses to the staff's questions supplied in their letter of October 10, 1978 serves as a basis for revising the Technical Specifications for Browns Ferry Units 1, 2 and 3. The staff believes that Note 7 for Table 3.2.B of the Browns Ferry Technical Specifications should be modified to read:

"7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150% above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150% above maximum steady state flow while also ensuring the initiation of isolation following a postulated break."

The licensee agreed via telephone on May 15, 1978 that this change would be acceptable.

The staff concludes that if the Browns Ferry Technical Specifications are changed as above, there is no decrease in any safety margin, no unreviewed safety questions as defined by 10 CFR 50.59 and the proposed changes to the Technical Specifications 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 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: March 11, 1981.

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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. 70 to Facility Operating License No. DPR-33, Amendment No. 66 to Facility Operating License No. DPR-52, and Amendment No. 42 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, Unit Nos. 1, 2 and 3, (the facility) located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

These amendments change the Technical Specifications to revise a note regarding the acceptance criteria in the startup test instructions for the High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems.

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

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§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 March 22, 1978, as supplemented by letter dated October 10, 1978, (2) Amendment No. 70 to License No. DPR-33, Amendment No. 66 to License No. DPR-52, and Amendment No. 42 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 11th day of March 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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