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

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Posted

Am-78
to DPR-52

February 1, 1982

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

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

Dear Mr. Parris:

The Commission has issued the enclosed Amendment Nos. 81, 78 and 50 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 consist of changes to the Technical Specifications in response to your request of October 19, 1981 (BFNP TS 168).

These amendments revise the Technical Specifications to increase the interval of time during which a primary containment integrated leak rate test must be performed from 3 1/3 years ± 8 months to 3 1/3 years ± 10 months.

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

Sincerely,

Richard J. Clark

Richard J. Clark, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 81 to DPR-33
2. Amendment No. 78 to DPR-52
3. Amendment No. 50 to DPR-68
4. Safety Evaluation
5. Notice

cc: w/enclosures
See next page

Mr. Hugh G. Parris

cc:

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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. 81
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 October 19, 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. 81, 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

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 1, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 81

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET No. 50-259

Revise Appendix A as follows:

1. Replace the following pages with the identically numbered pages:

   231
   274

   Marginal lines on the above pages indicate the areas being revised.

2. The overleaf pages are not being revised and should be retained.

Primary Containment

4.7.A Primary Containment

1. Prior to initial unit operation.

2. At approximately three and one-third year intervals so that any ten-year interval would include four ILRT's. These intervals may be extended up to ten months if necessary to coincide with refueling outage.

f. Except for the initial ILRT, all ILRT's shall be performed without leak repairs immediately prior to or during the test. If leak repairs are necessary in order to perform ILRT, they shall be preceded by local leak measurements where possible. The leak rate difference prior to and after repair shall be added to final integrated leak rate results, L_{pm} or L_{tm}. Following each ILRT, if the measured leak rate exceeds L_a, the condition shall be corrected. Following repairs, the integrated leak rate test need not be repeated provided local leakage rate measurements before and after repair demonstrate that the leakage rate reduction achieved by repairs reduces the overall measured integrated leak rate to an acceptable value.

g. Local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves at not less than 49.6 psig (except for the main steam isolation valves, see 4.7.A.i) each operafrom the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate (49 psig Method) or the allowable test leak rate (25 psig Method) by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on the NRC guide for developing leak rate testing and surveillance of reactor containment vessels. Allowing the test intervals to be extended up to 10 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

The primary containment is normally slightly pressurized during period of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, determining the oxygen concentration twice a week serves as an added assurance that the oxygen concentration will not exceed 4%.

3.7.B/3.7.C Standby Gas Treatment System and Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

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. 78
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 October 19, 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. 78, 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

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 1, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 78

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

1. Replace the following pages with the identically numbered pages:

   231
   274

   Marginal lines on the above pages indicate the areas being revised.

2. The overleaf pages are not being revised and should be retained.

Primary Containment

4.7.A Primary Containment

1. Prior to initial unit operation.

2. At approximately three and one-third year intervals so that any ten-year interval would include four ILRT's. These intervals may be extended up to ten months if necessary to coincide with refueling outage.

f. Except for the initial ILRT, all ILRT's shall be performed without leak repairs immediately prior to or during the test. If leak repairs are necessary in order to perform ILRT, they shall be preceded by local leak measurements where possible. The leak rate difference prior to and after repair shall be added to final integrated leak rate results, L_{om} or L_{tm}. Following each ILRT, if the measured leak rate exceeds L_a, the condition shall be corrected. Following repairs, the integrated leak rate test need not be repeated provided local leakage rate measurements before and after repair demonstrate that the leakage rate reduction achieved by repairs reduces the overall measured integrated leak rate to an acceptable value.

g. Local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves at not less than 49.6 psig (except for the main steam isolation valves, see 4.7.A.i) each operafrom the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate (49 psig Method) or the allowable test leak rate (25 psig Method) by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on the NRC guide for developing leak rate testing and surveillance of reactor containment vessels. Allowing the test intervals to be extended up to 10 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

The primary containment is normally slightly pressurized during period of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, determining the oxygen concentration twice a week serves as an added assurance that the oxygen concentration will not exceed 4%.

3.7.B/3.7.C Standby Gas Treatment System and Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

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. 50
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 October 19, 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. 50, 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

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment;
Changes to the Technical
Specifications

Date of Issuance: February 1, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

   237
   291

2. Marginal lines on the above pages indicate the areas being changed.

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

three and one-third year intervals so that any ten-year interval would include four ILRT's. These intervals may be extended up to ten months if necessary to coincide with refueling outage.

f. Except for the initial ILRT, all ILRT's shall be performed without leak repairs immediately prior to or during the test. If leak repairs are necessary in order to perform ILRT, they shall be preceded by local leak measurements where possible. The leak rate difference prior to and after repair shall be added to final integrated leak rate results, L_{pm} or L_{tm}. Following each ILRT, if the measured leak margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on the NRC guide for developing leak rate testing and surveillance of reactor containment vessels. Allowing the test intervals to be extended up to 10 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

The primary containment is normally slightly pressurized during period of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, determining the oxygen concentration twice a week serves as an added assurance that the oxygen concentration will not exceed 4%.

3.7.B/3.7.C <u>Standby Gas Treatment System and Secondary Containment</u>

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. All three standby gas treatment system fans are designed to automatically start upon containment isolation and to maintain the reactor building pressure to the design negative pressure so that all leakage should be in-leakage.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 50 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 October 19, 1981 (TVA BFNP TS 168), the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended 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 and revised Technical Specifications would increase the interval of time during which a primary containment integrated leak rate test (ILRT) must be performed from 3 1/3 years ± 8 months to 3 1/3 years ± 10 months. The proposed change would conform the requirement in the Browns Ferry Technical Specifications to the requirement in the BWR Standard Technical Specifications.

2.0 Evaluation

Section 4.7.A.2.e of the Browns Ferry Technical Specifications now requires that "primary containment ILRT's shall be performed at approximately three and one-third year intervals so that any ten-year interval would include four ILRTs. These intervals may be extended up to eight months if necessary to coincide with refueling outages." The proposed change to the Technical Specifications is to increase the extension period for performing the ILRTs from eight months to ten months. In order to complete NRC required modifications (e.g., TMI Lessons Learned, Mark I Torus Upgrade, etc.), recent outages at Browns Ferry have been longer than projected; the 1981 outage for Browns Ferry Unit 1 was 6 months vs a projected outage of 3 1/2 months when the facility shutdown on April 10, 1981. This in turn has altered the projected dates for future refueling outages.

The BWR Standard Technical Specifications, NUREG-0123, Revision 3, dated fall 1980, in Section 4.6.1.2.a states: "Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 $\pm$ 10 month intervals during shutdown at either P_a, (40.4) psig, or at $\overline{P}_t$, (20.2) psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection."

The proposed change in the Browns Ferry Technical Specifications would conform them to the above requirement in the BWR Standard Technical Specifications. This change only allows ILRT intervals to be extended an additional 2 months if necessary and does not change or effect any safety system or other requirements on how the ILRTs must be conducted. Four ILRTs must still be performed in any 10-year period as required by 10 CFR Part 50.

We conclude that the proposed change is acceptable on the bases for this test interval in the BWR Standard Technical Specifications.

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: February 1, 1982

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. 81 to Facility Operating License No. DPR-33, Amendment No. 78 to Facility Operating License No. DPR-52, and Amendment No. 50 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.

The amendments revise the Technical Specifications to increase the interval of time during which a primary containment integrated leak rate test must be performed from 3 1/3 years ± 8 months to 3 1/3 years ± 10 months.

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 October 19, 1981, (2) Amendment No. 81 to License No. DPR-33, Amendment No. 78 to License No. DPR-52, and Amendment No. 50 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 1st day of February 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing