Docket Nos. 50-259/260/296

December 27, 1983

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. 94 , 87 and 60 to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3. These amendments are in response to your application dated March 25, 1983 (TVA BFNP TS 186) as modified with the concurrence of your staff.

The amendments change the Technical Specifications to add more stringent requirements to Section 3.6.C on allowable primary coolant leakage into the drywell.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next monthly Federal Register notice.

Sincerely,

Original signed by/

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

Enclosures:
1. Amendment No. 94 to
   License No. DPR-33
2. Amendment No. 87 to
   License No. DPR-52
3. Amendment No. 60 to
   License No. DPR-68
4. Safety Evaluation

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cc w/enclosures:
See next page

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Mr. Hugh G. Parris
Tennessee Valley Authority
Browns Ferry Nuclear Plant, Units 1, 2 and 3

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 1

## AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 25, 1983, 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 Operating 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. 94, 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 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: December 27, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 94

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

   180

   181

   219

2. The marginal lines on these pages denote the area being changed.

3.6 PRIMARY SYSTEM BOUNDARY

C. Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25

b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24 hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below. Drywell leakage shall be measured and recorded every 8 hours.

c. During the first 24 hours in the RUN mode following startup, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

4.6 PRIMARY SYSTEM BOUNDARY

C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per day.

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

3.6.C Coolant Leakage

2. Both the sump and air sampling systems shall be operable during reactor power operation. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 72 hours.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

D. Relief Valves

1. When more than one relief valves are known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours.

E. Jet Pumps

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

4.6.C Coolant Leakage

D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. Once during each operating cycle, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3. The integrity of the relief/safety valve bellows shall be continuously monitored.

4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup or run modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

   a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

detected reasonably in a matter of few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the unit should be shut down to allow further investigation and corrective action.

The 2 gpm limit for coolant leakage rate increase over any 24 hour period is a limit specified by the NRC (Reference 2.) This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

REFERENCE

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

## 3.6.D/4.6.D Relief Valves

To meet the safety basis thirteen relief valves have been installed on the unit with a total capacity of 83.9% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves operable, results in adequate margin to the code allowable overpressure limit of 1375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1375 psig.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 25, 1983, 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 Operating 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. 87, 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 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: December 27, 1983

ATTACHMENT TO LICENSE AMENDMENT NO.87

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

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

   180

   181

   219

2. The marginal lines on these pages denote the area being changed.

## 3.6 PRIMARY SYSTEM BOUNDARY

### C. Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 –

   b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24 hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below. Drywell leakage shall be measured and recorded every 8 hours.

   c. During the first 24 hours in the RUN mode following startup, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

## 4.6 PRIMARY SYSTEM BOUNDARY

### C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per day.

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

3.6.C Coolant Leakage

2. Both the sump and air sampling systems shall be operable during reactor power operation. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 72 hours.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

D. Relief Valves

1. When more than one relief valves are known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours.

E. Jet Pumps

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

4.6.C Coolant Leakage

D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. Once during each operating cycle, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3. The integrity of the relief/safety valve bellows shall be continuously monitored.

4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup or run modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

   a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

detected reasonably in a matter of few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the unit should be shut down to allow further investigation and corrective action.

The 2 gpm limit for coolant leakage rate increase over any 24 hour period is a limit specified by the NRC (Reference 2.) This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

REFERENCE

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

## 3.6.D/4.6.D Relief Valves

To meet the safety basis thirteen relief valves have been installed on the unit with a total capacity of 83.9% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves operable, results in adequate margin to the code allowable overpressure limit of 1375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1375 psig.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 60
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 25, 1983, 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 Operating 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. 60, 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 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: December 27, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 60

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

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

   191

   192

   224

2. The marginal lines on these pages denote the area being changed.

3.6 PRIMARY SYSTEM BOUNDARY

C. Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24 hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.

c. During the first 24 hours in the RUN mode following startup, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

4.6 PRIMARY SYSTEM BOUNDARY

C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per day.

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

3.6 PRIMARY SYSTEM BOUNDARY

2. Both the sump and air sampling sampling systems shall be operable during reactor power operation. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding seven days.

   The air sampling system may be removed from service for a period of 4 hours for calibration, functional testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

D. Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours.

4.6 PRIMARY SYSTEM BOUNDARY

D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. Once during each operating cycle, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3. At least one relief valve shall be disassembled and inspected each operating cycle.

limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5-gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the unit should be shutdown to allow further investigation and corrective action.

The 2 gpm limit for coolant leakage rate increase over any 24 hour period is a limit specified by the NRC (Reference 2). This limit applies only during the RUN mode to avoid being penalized for the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

## REFERENCES

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)
2. Safety Evaluation Report (SER) on IE Bulletin 82-03

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 60 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 March 25, 1983 (TVA BFNP TS 186) the Tennessee Valley Authority (the licensee or TVA) requested amendments to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3. The application by TVA was in response to a request by the NRC staff on March 11, 1983 to provide revised Technical Specifications for Browns Ferry Unit 2 with more stringent requirements on unidentified leakage in the drywell. The requested changes were the same as those in the BWR Standard Technical Specifications. During the refueling and modification outage of Unit 2, which extended from July 30, 1982 to March 18, 1983, TVA found indications of cracks in two of the ten sweep-o-let to manifold welds in the recirculation system. TVA proposed to operate in Cycle 5 with these two indications. We performed an independent materials and fracture mechanics evaluation and concluded that operation throughout the next cycle with these indications was acceptable but that certain additional compensatory measures were warranted, such as more stringent requirements on unidentified leakage.

## 2.0 Discussion

The staff requested TVA to submit a change to the Technical Specifications limiting the rate of increase for unidentified drywell leakage to 2 gallons per minute (gpm) in a 24-hour period. This is the same requirement that is in the BWR Standard Technical Specifications (NUREG-0123, Rev. 3). The requested change was submitted by TVA's letter of March 25, 1983. On March 4, 1983, IE Bulletin 83-02 was issued requiring augmented inservice inspection of recirculation and residual heat removal system piping for BWRs shutting down for refueling after February 1983.

Inspections performed in accordance with this Bulletin revealed indications of pipe cracks in most facilities. As a result, the staff concluded that even more enhanced surveillance of possible leakage was warranted. Specifically, the staff proposed that the frequency for checking the leakage rate be increased once per day to once per shift and that the allowable period for plant operation without the leakage monitoring systems in operation be reduced from 7 to 3 days. TVA subsequently imposed these limits administratively on all three Browns Ferry units. Similar limits on unidentified leakage have been incorporated in the Technical Specifications for numerous BWRs during the past year. The NRC staff proposed these additional surveillance requirements to TVA as a supplement to the March 25, 1983 submittal. The additional changes were accepted by the TVA staff, and, as noted above, were administratively imposed voluntarily by TVA. Thus, the changes to the Technical Specifications encompassed by these amendments are already in effect. However, since Browns Ferry Unit 1 will be returning to power in Cycle 6 in the near future with nine unrepaired welds, the staff has determined that the changes should be incorporated in the Technical Specifications by amendments.

## 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 §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) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

Principal Contributors: W. Hazelton, W. Koo and R. Clark

Dated: December 27, 1983