April 9, 1993

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

Tennessee Valley Authority
ATTN: Dr. Mark O. Medford, Vice President
Nuclear Assurance, Licensing & Fuels
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Medford:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M84491, M84492, AND M84493) (TS 323)

The Commission has issued the enclosed Amendment Nos.193, 208 , and 165 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, respectively, in response to an application dated September 10, 1992, by the Tennessee Valley Authority (TVA). These amendments revise the BFN Technical Specifications (TS) by restoring the operability requirements of the Control Room Emergency Ventilation System (CREVS) that were temporarily modified by previous license amendments issued on September 18, 1989. The enclosed license amendments also remove a list of dampers in accordance with Generic Letter 91-08, and revise associated Bases.

A copy of the staff's Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

~~Frederick J. Hebdon~~, Director for
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.193to License No. DPR-33
2. Amendment No.208to License No. DPR-52
3. Amendment No.165to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

| LA:PDII-4 | PM:PDII-4 | PM:PDII-4 | OGC | D:PDII-4 | PRPB |
|---|---|---|---|---|---|
| for MSanders | TRoss:sl | JWilliams | Sturk | FHebdon for | LCunningham |
| 4/7/93 | 4/8/93 | 4/8/93 | 4/9/93 | 4/9/93 | 4/9/93 |

DOCUMENT NAME:BFN94491.AMD

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AMENDMENT NO. 193 FOR BROWNS FERRY UNIT 1 - DOCKET NO. 50-259
AMENDMENT NO. 208 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260
AMENDMENT NO. 165 FOR BROWNS FERRY UNIT 3 - DOCKET NO. 50-296
DATED: April 9, 1993

DISTRIBUTION:
Docket File
NRC PDR & Local PDR
BFN Reading File
S. Varga 14-E-4
G. Lainas 14-H-3
F. Hebdon
M. Sanders
T. Ross
J. Williams
P. Frederickson RII
OGC 15-B-13
D. Hagan MNBB-3302
E. Jordan MNBB-3302
G. Hill P1-130 (2 per docket)
Wanda Jones MNBB-7103
C. Grimes 11-E-21
E. Lee
ACRS (10)
OPA 2-G-5
OC/LFDCB MNBB-9112

cc: Plant Service List

Tennessee Valley Authority
ATTN: Dr. Mark O. Medford



cc:
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Tennessee Valley Authority
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 193
License No. DPR-33

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 10, 1992, 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, 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. 193, 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 9, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 193

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf* and spillover** pages are provided to maintain document completeness.

| REMOVE | INSERT |
|---|---|
| 3/7/4.7-19 | 3.7/4.7-19 |
| 3.7/4.7-20 | 3.7/4.7-20 |
| 3.7/4.7-35 | 3.7/4.7-35* |
| 3.7/4.7-36 | 3.7/4.7-36 |

LIMITING CONDITIONS FOR OPERATION

3.7.E. Control Room Emergency Ventilation

1. Except as specified in Specification 3.7.E.3 below, both control room emergency pressurization systems shall be OPERABLE at all times when any reactor vessel contains irradiated fuel.

2. a. The results of the inplace cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

   b. The results of laboratory carbon sample analysis shall show $\geq$90% radioactive methyl iodide removal at a velocity when tested in accordance with ASTM D3803 (130°C, 95% R.H.).

SURVEILLANCE REQUIREMENTS

4.7.E Control Room Emergency Ventilation

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate ($\pm$ 10%).

2. a. The tests and sample analysis of Specification 3.7.E.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

   b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

LIMITING CONDITIONS FOR OPERATION

3.7.E. Control Room Emergency Ventilation

c. System flow rate shall be shown to be within ±10% design flow when tested in accordance with ANSI N510-1975.

3. From and after the date that one of the control room emergency pressurization systems is made or found to be inoperable for any reason, REACTOR POWER OPERATIONS or refueling operations are permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE.

4. If these conditions cannot be met, reactor shutdown shall be initiated and all reactors shall be in COLD SHUTDOWN within 24 hours for REACTOR POWER OPERATIONS and refueling operations shall be terminated within 2 hours.

SURVEILLANCE REQUIREMENTS

4.7.E. Control Room Emergency Ventilation

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

d. Each circuit shall be operated at least 10 hours every month.

3. At least once every 18 months, automatic initiation of the control room emergency pressurization system shall be demonstrated.

4. During the simulated automatic actuation test of this system (see Table 4.2.G), it shall be verified that the necessary dampers operate as required.

in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low-level isolation is provided.

Groups 4 and 5 - Process lines are designed to remain OPERABLE and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of Groups 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - Lines are connected to the primary containment but not directly to the reactor vessel. These valves are isolated on reactor low water level (538"), high drywell pressure, or reactor building ventilation high radiation which would indicate a possible accident and necessitate primary containment isolation.

Group 7 - Process lines are closed only on the respective turbine steam supply valve not fully closed. This assures that the valves are not open when HPCI or RCIC action is required.

Group 8 - Line (traveling in-core probe) is isolated on high drywell pressure or reactor low water level (538"). This is to assure that this line does not provide a leakage path when containment pressure or reactor water level indicates a possible accident condition.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent, an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

These valves are highly reliable, have low service requirements and most are normally closed. The initiating sensors and associated trip logic are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of $1.1 \times 10^{-7}$ that a line will not isolate. More frequent testing for valve OPERABILITY in accordance with Specification 1.0.MM results in a greater assurance that the valve will be OPERABLE when needed.

The main steam line isolation valves are functionally tested per Specification 1.0.MM to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25-inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

### 3.7.E/4.7.E Control Room Emergency Ventilation

The control room emergency ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room emergency ventilation system is designed to automatically start upon control room isolation and to assist other sources of pressurization in maintaining the control room at a positive pressure.

High efficiency particulate absolute (HEPA) filters are installed prior to the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation or refueling operation may continue for a

UNITED STATES
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WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 208
License No. DPR-52

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 10, 1992, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 208 , 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director for
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 9, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 208

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf* and spillover** pages are provided to maintain document completeness.

| REMOVE | INSERT |
|---|---|
| 3.7/4.7-19 | 3.7/4.7-19 |
| 3.7/4.7-20 | 3.7/4.7-20 |
| 3.7/4.7-35 | 3.7/4.7-35* |
| 3.7/4.7-36 | 3.7/4.7-36 |

LIMITING CONDITIONS FOR OPERATION

3.7.E. Control Room Emergency Ventilation

1. Except as specified in Specification 3.7.E.3 below, both control room emergency pressurization systems shall be OPERABLE at all times when any reactor vessel contains irradiated fuel.

2. a. The results of the inplace cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

   b. The results of laboratory carbon sample analysis shall show $\geq$90% radioactive methyl iodide removal at a velocity when tested in accordance with ASTM D3803 (130°C, 95% R.H.).

SURVEILLANCE REQUIREMENTS

4.7.E Control Room Emergency Ventilation

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to to be less than 6 inches of water at system design flow rate ($\pm$ 10%).

2. a. The tests and sample analysis of Specification 3.7.E.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

   b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

LIMITING CONDITIONS FOR OPERATION

3.7.E. Control Room Emergency Ventilation

c. System flow rate shall be shown to be within ±10% design flow when tested in accordance with ANSI N510-1975.

3. From and after the date that one of the control room emergency pressurization systems is made or found to be inoperable for any reason, REACTOR POWER OPERATIONS or refueling operations are permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE.

4. If these conditions cannot be met, reactor shutdown shall be initiated and all reactors shall be in COLD SHUTDOWN within 24 hours for REACTOR POWER OPERATIONS and refueling operations shall be terminated within 2 hours.

SURVEILLANCE REQUIREMENTS

4.7.E. Control Room Emergency Ventilation

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

d. Each circuit shall be operated at least 10 hours every month.

3. At least once every 18 months, automatic initiation of the control room emergency pressurization system shall be demonstrated.

4. During the simulated automatic actuation test of this system (see Table 4.2.G), it shall be verified that the necessary dampers operate as required.

in the system, isolation is provided by high temperature in the cleanup system area or high drain temperature. Also, since the vessel could potentially be drained through the cleanup system, a low-level isolation is provided.

Groups 4 and 5 - Process lines are designed to remain OPERABLE and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of Groups 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - Lines are connected to the primary containment but not directly to the reactor vessel. These valves are isolated on reactor low water level (538"), high drywell pressure, or reactor building ventilation high radiation which would indicate a possible accident and necessitate primary containment isolation.

Group 7 - (Deleted)

Group 8 - Line (traveling in-core probe) is isolated on high drywell pressure or reactor low water level (538"). This is to assure that this line does not provide a leakage path when containment pressure or reactor water level indicates a possible accident condition.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent, an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

These valves are highly reliable, have low service requirements and most are normally closed. The initiating sensors and associated trip logic are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of $1.1 \times 10^{-7}$ that a line will not isolate. More frequent testing for valve OPERABILITY in accordance with Specification 1.0.MM results in a greater assurance that the valve will be OPERABLE when needed.

The main steam line isolation valves are functionally tested per Specification 1.0.MM to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25-inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

3.7.E/4.7.E Control Room Emergency Ventilation

The control room emergency ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room emergency ventilation system is designed to automatically start upon control room isolation and to assist other sources of pressurization in maintaining the control room at a positive pressure.

High efficiency particulate absolute (HEPA) filters are installed prior to the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation or refueling operation may continue for a

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 165
License No. DPR-68

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 10, 1992, 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. 165, 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director for
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: **April 9, 1993**

ATTACHMENT TO LICENSE AMENDMENT NO. 165

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf* and spillover** pages are provided to maintain document completeness.

| REMOVE | INSERT |
|---|---|
| 3.7/4.7-19 | 3.7/4.7-19 |
| 3.7/4.7-20 | 3.7/4.7-20 |
| 3.7/4.7-34 | 3.7/4.7-34* |
| 3.7/4.7-35 | 3.7/4.7-35 |

LIMITING CONDITIONS FOR OPERATION

3.7.E. Control Room Emergency Ventilation

1. Except as specified in Specification 3.7.E.3 below, both control room emergency pressurization systems shall be OPERABLE at all times when any reactor vessel contains irradiated fuel.

2. a. The results of the inplace cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

   b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity when tested in accordance with ASTM D3803 (130°C, 95% R.H.).

SURVEILLANCE REQUIREMENTS

4.7.E Control Room Emergency Ventilation

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to to be less than 6 inches of water at system design flow rate (± 10%).

2. a. The tests and sample analysis of Specification 3.7.E.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

   b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| 3.7.E. Control Room Emergency Ventilation | 4.7.E. Control Room Emergency Ventilation |
| c. System flow rate shall be shown to be within ±10% design flow when tested in accordance with ANSI N510-1975. | c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing. |
| | d. Each circuit shall be operated at least 10 hours every month. |
| 3. From and after the date that one of the control room emergency pressurization systems is made or found to be inoperable for any reason, REACTOR POWER OPERATIONS or refueling operations are permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE. | 3. At least once every 18 months, automatic initiation of the control room emergency pressurization system shall be demonstrated. |
| 4. If these conditions cannot be met, reactor shutdown shall be initiated and all reactors shall be in COLD SHUTDOWN within 24 hours for REACTOR POWER OPERATIONS and refueling operations shall be terminated within 2 hours. | 4. During the simulated automatic actuation test of this system (see Table 4.2.G), it shall be verified that the necessary dampers operate as required. |

in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low-level isolation is provided.

Groups 4 and 5 - Process lines are designed to remain OPERABLE and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of Groups 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - Lines are connected to the primary containment but not directly to the reactor vessel. These valves are isolated on reactor low water level (538"), high drywell pressure, or reactor building ventilation high radiation which would indicate a possible accident and necessitate primary containment isolation.

Group 7 - Process lines are closed only on the respective turbine steam supply valve not fully closed. This assures that the valves are not open when HPCI or RCIC action is required.

Group 8 - Line (traveling in-core probe) is isolated on high drywell pressure or reactor low water level (538"). This is to assure that this line does not provide a leakage path when containment pressure or reactor water level indicates a possible accident condition.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent, an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

These valves are highly reliable, have low service requirements and most are normally closed. The initiating sensors and associated trip logic are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of $1.1 \times 10^{-7}$ that a line will not isolate. More frequent testing for valve OPERABILITY in accordance with Specification 1.0.MM results in a greater assurance that the valve will be OPERABLE when needed.

The main steamline isolation valves are functionally tested per Specification 1.0.MM to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25-inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

3.7.E/4.7.E Control Room Emergency Ventilation

The control room emergency ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room emergency ventilation system is designed to automatically start upon control room isolation and to assist other sources of pressurization in maintaining the control room at a positive pressure.

High efficiency particulate absolute (HEPA) filters are installed prior to the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation or refueling operation may continue for a

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE 4

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NO. 193 TO FACILITY OPERATING LICENSE NO. DPR-33

# AMENDMENT NO. 208 TO FACILITY OPERATING LICENSE NO. DPR-52

# AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. DPR-68

# TENNESSEE VALLEY AUTHORITY

# BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

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

## 1.0 INTRODUCTION

By letter dated September 10, 1992, the Tennessee Valley Authority (TVA, the licensee) submitted a request for changes to the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3 Technical Specifications (TS). The requested changes would restore the TS operability requirements of the Control Room Emergency Ventilation System (CREVS) that were temporarily modified by previous license amendments issued on September 18, 1989. The proposed TS changes would also remove the detailed list of dampers required to operate during automatic actuation of CREVS, in accordance with Generic Letter (GL) 91-08, "Removal of Component Lists from Technical Specifications." Furthermore, TVA would revise the associated TS Bases for CREVS to delete the description of the temporary changes that were in place only for Unit 2 Cycle 6, and to reflect the new CREVS design.

## 2.0 EVALUATION

By letter dated September 18, 1989, the NRC issued license amendments for BFN, Units 1, 2, and 3, that temporarily revised the TS to allow for fuel movement and reactor operation with the CREVS considered inoperable. These temporary TS would be in force until just prior to startup of Unit 2 for Cycle 7 operations. However, during Cycle 6, the CREVS was required to meet all applicable TS surveillances and associated action statements.

The staff considered CREVS inoperable during Unit 2 Cycle 6 only because it did not meet its design basis for pressurizing the Control Bay Habitability Zone (CBHZ) with essentially zero unfiltered inleakage as described by the BFN Updated Final Safety Analysis Report (UFSAR). TVA had discovered, prior to startup of Unit 2 for Cycle 6, that several thousand cubic feet per minute (CFM) of potentially contaminated air during a postulated accident could bypass CREVS and enter the CBHZ as unfiltered inleakage. By virtue of the license amendments issued September 18, 1989, the staff allowed TVA one fuel

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cycle of Unit 2 operation to effect appropriate corrective actions that would restore CREVS operability.

By letters dated May 5, 1992, July 31, 1992, and March 1, 1993, TVA submitted the details of its corrective action plan for CREVS. Upon implementation of these corrective actions during the Unit 2 Cycle 6 refueling outage, TVA stated that BFN would be in full compliance with General Design Criterion (GDC) 19, "Control Room," of 10 CFR Part 50, Appendix A. The staff's safety evaluation (SE) of TVA's corrective action plan for CREVS will be issued under separate correspondence prior to restart of Unit 2 for Cycle 7 operations. The staff's SE contained herein, merely addresses the adequacy of TVA's proposed TS changes for CREVS as submitted by TVA letter dated September 10, 1992.

TVA proposed changes to Limiting Conditions for Operations (LCO) 3.7.E.1, 3.7.E.3, and 3.7.E.4 to remove the temporary TS changes that were in place for Unit 2 Cycle 6 only. These temporary TS changes allowed for fuel movement and reactor operation even with CREVS inoperable. However, once Unit 2 restarts for Cycle 7 operations these temporary TS changes are no longer applicable. The removal of the expired TS requirements is principally an editorial change. Consequently, the staff considers TVA's proposed changes acceptable. However, in accepting the proposed changes to TS LCOs 3.7.E.1, 3.7.E.3, and 3.7.E.4, and deletion of the associated footnote, the staff is not indicating that the CBHZ design at BFN has been restored to zero unfiltered in-leakage. As described in TVA's corrective action plan for CREVS, a substantial amount of unfiltered in-leakage is now assumed as part of the new design basis. TVA has stated that the quantity of unfiltered in-leakage entering the BFN CBHZ does not result in doses to the control room operators in excess of GDC 19. The NRC staff is currently reviewing TVA's corrective action plan for CREVS and its revised control room operator dose calculations for confirming compliance with GDC 19. NRC acceptance of the proposed TS changes should not be misconstrued as staff approval of TVA's new CBHZ design basis or corrective action plan for CREVS.

TVA also proposed to revise the Bases for TS 3.7.E/4.7.E. These Bases would be revised to remove the description of the temporary TS changes discussed above. In addition, they would reflect the new CBHZ design basis that allows for some in-leakage, as opposed to the original UFSAR design basis that specifically stated all leakage would be out-leakage. The staff reviewed TVA's proposed Bases and finds them acceptable. However, NRC acceptance of the revised design basis for CREVS, as described in TVA's proposed Bases for TS 3.7.E/4.7.E, is contingent upon the staff also accepting TVA's corrective action plan for CREVS. Although the staff recognizes that some in-leakage is inevitable under certain accident conditions, the staff is reviewing TVA's assumptions and calculations regarding the quantity of in-leakage. Should the staff conclude that TVA's corrective action plan for CREVS or dose calculations are unacceptable, an additional Bases revision may be necessary.

Lastly, TVA proposed to remove the list of dampers from TS Surveillance Requirement (SR) 4.7.E.4. TVA stated in its September 10, 1992 letter, that these dampers were included in the control room isolation and pressurization functional test procedure for BFN. This test procedure is a TVA controlled procedure subject to the change control provisions of the Administrative Controls section of TS (i.e., Section 6.8.1.1.). The guidelines of GL 91-08 establish an acceptable alternative to identifying lists of specific components in the TS. The staff reviewed TVA's proposed TS change to delete the list of dampers from SR 4.7.E.4 against the guidance of GL 91-08. Based on this review, the staff considers this change acceptable.

During its review of this amendment application the staff noticed that TVA did not propose the additional surveillance requirements for testing unfiltered CBHZ in-leakage as deemed necessary by the staff in its SE dated September 18, 1989. By letter dated March 1, 1993, TVA addressed the staff's concern regarding the necessity of a TS SR to demonstrate the unfiltered in-leakage rate. In this letter TVA committed to establish a Surveillance Instruction (SI) that would determine the CBHZ in-leakage rate every cycle. However, TVA maintained that a change to the BFN TS to explicitly prescribe an SR to measure CBHZ in-leakage was not necessary, and inconsistent with the NRC's Improved Standard Technical Specifications (ISTS). The staff reviewed TVA's justifications, but is not convinced that the new CREVS design basis (still under staff review) will not warrant additional TS requirements (e.g., SR for CBHZ unfiltered in-leakage) per 10 CFR 50.36.

The staff acknowledges TVA's commitment to perform measurements of the CBHZ unfiltered in-leakage each cycle as part of a BFN SI. However, the issue regarding adequate surveillance requirements for determining CBHZ integrity will be addressed by the staff's SE on TVA's corrective action plan for CREVS to be issued prior to Unit 2 Cycle 7 startup.

## 3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendment. The State official had no comments.

## 4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (57 FR 48829). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: T. Ross

Date: April 9, 1993