March 18, 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 TECHNICAL SPECIFICATION AMENDMENTS (TAC NOS. M84639, M84640 AND M84641) (TS 330)

The Commission has issued the enclosed Amendment Nos. 191 , 206 , and 163 , to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2, and 3, respectively. These amendments are in response to your application, dated September 28, 1992, to remove lists of certain circumferential pipe welds from Technical Specifications in accordance with Generic Letter 91-08.

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

Sincerely,

Original signed by
Victor Nerses for

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

Enclosures:
1. Amendment No.191 to License No. DPR-33
2. Amendment No.206 to License No. DPR-52
3. Amendment No.163 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

OFC	PDII-4/LA	PDII-4/PE	PDII-4/PM	PDII-4/PM	OGC NLO	PDII-4
NAME	MSanders	CBeardslee:jkd	TRoss	JWilliams	MZOBLER	FHebdon
DATE	12/30/92	12/30/92	3/15/93	3/18/93	3/10/93	/ /93

3/5/93 ms

One minor Editorial correction

DF01 / 1/1 CP 1

Tennessee Valley Authority
ATTN: Dr. Mark O. Medford

Browns Ferry Nuclear Plant

cc:
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Site Quality Manager
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AMENDMENT NO. 191 FOR BROWNS FERRY UNIT 1 - DOCKET NO. 50-259
AMENDMENT NO. 206 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260
AMENDMENT NO. 163 FOR BROWNS FERRY UNIT 3 - DOCKET NO. 50-296
DATED: March 18, 1993

DISTRIBUTION:

Docket File	
NRC PDR	
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G. Hill(6)	P1-137
Wanda Jones	MNBB-7103
C. Grimes	
ACRS(10)	
OPA	2-G-5
OC/LFDCB	MNBB-91122

cc: Plant Service List

250004

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. 191
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 September 28, 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-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. 191, 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

Victor Nerses for

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: March 18, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 191

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
3.6/4.6-13	3.6/4.6-13
3.6/4.6-14	3.6/4.6-14
3.6/4.6-32	3.6/4.6-32*
3.6/4.6-33	3.6/4.6-33

LIMITING CONDITIONS FOR OPERATION

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.

   a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

   b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.

SURVEILLANCE REQUIREMENTS

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

4.6.G. Structural Integrity

3. For Unit 1 an augmented inservice surveillance program shall be performed to monitor potential corrosive effects of chloride residue released during the March 22, 1975 fire. The augmented inservice surveillance program is specified as follows:

   a. Browns Ferry Mechanical Maintenance Instruction 53, dated September 22, 1975, paragraph 4, defines the liquid penetrant examinations required during the first, second, third and fourth refueling outages following the fire restoration.

   b. Browns Ferry Mechanical Maintenance Instruction 46, dated July 18, 1975. Appendix B, defines the liquid penetrant examinations required during the sixth refueling outage following the fire restoration.

3.6.E/4.6.E (Cont'd)

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Steady-state operation without forced recirculation will not be permitted for more than 12 hours. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50% of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

3.6.G/4.6.G (Cont'd)

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in plant procedures to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

An augmented inservice surveillance program is required to determine whether any stress corrosion has occurred in any stainless steel piping, stainless components, and highly-stressed alloy steel such as hanger springs, as a result of environmental conditions associated with the March 22, 1975 fire.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)

2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code

3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)

4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

5. Mechanical Maintenance Instruction 46 (Mechanical Equipment, Concrete, and Structural Steel Cleaning Procedure for Residue From Plant Fire - Units 1 and 2)

6. Mechanical Maintenance Instruction 53 (Evaluation of Corrosion Damage of Piping Components Which Were Exposed to Residue From March 22, 1975 Fire)

7. Plant Safety Analysis (BFNP FSAR Subsection 4.12)

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. 206
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 September 28, 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 206 , 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

Victor Nerses for

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: March 18, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 206

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
3.6/4.6-13	3.6/4.6-13
3.6/4.6-14	3.6/4.6-14
3.6/4.6-32	3.6/4.6-32*
3.6/4.6-33	3.6/4.6-33

LIMITING CONDITIONS FOR OPERATION

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.

   a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a COLD SHUTDOWN CONDITION or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

SURVEILLANCE REQUIREMENTS

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.G Structural Integrity 3.6.G.1 (Cont'd) b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.	

3.6.E/4.6.E (Cont'd)

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Operation without forced recirculation is permitted for up to 12 hours when the reactor is not in the RUN mode. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring at least one recirculation pump to be operable while in the RUN mode (i.e., requiring a manual scram if both recirculation pumps are tripped) provides protection against the potential occurrence of core thermal-hydraulic instabilities at low flow conditions.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50% of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6/4.6 BASES

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in plant procedures to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

An augmented inservice surveillance program is required to determine whether any stress corrosion has occurred in any stainless steel piping, stainless components, and highly-stressed alloy steel such as hanger springs, as a result of environmental conditions associated with the March 22, 1975 fire.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)

2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code

3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)

4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

5. Mechanical Maintenance Instruction 46 (Mechanical Equipment, Concrete, and Structural Steel Cleaning Procedure for Residue From Plant Fire - Units 1 and 2)

6. Mechanical Maintenance Instruction 53 (Evaluation of Corrosion Damage of Piping Components Which Were Exposed to Residue From March 22, 1975 Fire)

7. Plant Safety Analysis (BFNP FSAR Subsection 4.12)

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 163
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 September 28, 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. 163, 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

Victor Nerses for

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: March 18, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 163

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
3.6/4.6-13	3.6/4.6-13
3.6/4.6-14	3.6/4.6-14
3.6/4.6-32	3.6/4.6-32*
3.6/4.6-33	3.6/4.6-33

LIMITING CONDITIONS FOR OPERATION

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.

   a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

   b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.

SURVEILLANCE REQUIREMENTS

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

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3.6.E/4.6.E (Cont'd)

area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Steady-state operation without forced recirculation will not be permitted for more than 12 hours. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50 percent of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

3.6.G/4.6.G (Cont'd)

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in plant procedures to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

References

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)

2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code

3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)

4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

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

ENCLOSURE 4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 191 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 206 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 163 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

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

1.0 INTRODUCTION

By letter dated September 28, 1992, the Tennessee Valley Authority (TVA) submitted a request for changes to the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3 Technical Specifications (TS). The requested changes would relocate the list of additional inspections required to be performed on certain circumferential pipe welds from TS 4.6.G to plant procedures. Guidance on these proposed changes was provided to all holders of operating licenses or construction permits for nuclear power reactors by Generic Letter (GL) 91-08, "Removal of Component Lists From Technical Specification," dated May 6, 1991. By relocating the lists of pipe welds from TS to plant procedures, any subsequent changes to these particular component lists would be controlled pursuant to provisions of 10 CFR 50.59 and TS Section 6, "Administrative Controls."

2.0 EVALUATION

Tennessee Valley Authority proposed to remove the lists of certain circumferential pipe welds which required additional inspections, from Units 1, 2, and 3 TS Section 4.6.G.2. The contents of these lists will be incorporated into the applicable TVA program and procedures, which are subject to the administrative controls prescribed in Section 6.8, "Procedural/Instructions and Programs," of the BFN TS. As is required by GL 91-08, proposed TS Section 4.6.G.2 contains an appropriate description of the scope of the components to which the TS requirements apply.

The BASES for TS Section 3.6.G/4.6.G provides a more specific description of the type of pipe welds that are listed. Proposed changes to the BASES states that the list of circumferential pipe welds to be inspected is in plant procedures. Furthermore, at the request of NRC staff, TVA submitted additional information by letter dated February 22, 1993, regarding the technical basis for selecting and inspecting these welds.

After reviewing TVA's amendment application, the staff concludes that the proposed TS changes for BFN, Units 1, 2, and 3, are primarily administrative in nature and conform with the guidance of GL 91-08 and are therefore acceptable.

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

These amendments change a requirement with respect to installation or use and surveillance of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 55592). Accordingly, these 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: C. Beardslee and T. Ross

Date: March 18, 1993

March 18, 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 TECHNICAL SPECIFICATION AMENDMENTS (TAC NOS. M84639, M84640 AND M84641) (TS 330)

The Commission has issued the enclosed Amendment Nos. 191 , 206 , and 163 , to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2, and 3, respectively. These amendments are in response to your application, dated September 28, 1992, to remove lists of certain circumferential pipe welds from Technical Specifications in accordance with Generic Letter 91-08.

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

Sincerely,

Original signed by
Victor Nerses for

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

Enclosures:
1. Amendment No.191 to License No. DPR-33
2. Amendment No.206 to License No. DPR-52
3. Amendment No.163 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

OFC	PDII-4/LA	PDII-4/PE	PDII-4/PM	PDII-4/PM	OGC NLO	PDII-4
NAME	MSanders	CBeardslee:jkd	TRoss	JWilliams	MZOBLER	FHebdon
DATE	12/30/92	12/30/92	3/5/93	3/8/93	3/10/93	/ /93

3/5/93

One minor Editorial correction

9303290288

AMENDMENT NO. 191 FOR BROWNS FERRY UNIT 1 - DOCKET NO. 50-259
AMENDMENT NO. 206 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260
AMENDMENT NO. 163 FOR BROWNS FERRY UNIT 3 - DOCKET NO. 50-296
DATED: March 18, 1993

<u>DISTRIBUTION:</u>

Docket File	
NRC PDR	
Local PDR	
BFN Reading File	
S. Varga	14-E-4
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M. Sanders	
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J. Williams	
OGC	15-B-18
P. Kellogg	RII
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D. Hagan	MNBB-3302
G. Hill(6)	P1-137
Wanda Jones	MNBB-7103
C. Grimes	
ACRS(10)	
OPA	2-G-5
OC/LFDCB	MNBB-91122

cc: Plant Service List

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

March 18, 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 TECHNICAL SPECIFICATION AMENDMENTS (TAC NOS. M84639, M84640 AND M84641) (TS 330)

The Commission has issued the enclosed Amendment Nos. 191 , 206 , and 163 , to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2, and 3, respectively. These amendments are in response to your application, dated September 28, 1992, to remove lists of certain circumferential pipe welds from Technical Specifications in accordance with Generic Letter 91-08.

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

Sincerely,

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

Enclosures:
1. Amendment No. 191 to License No. DPR-33
2. Amendment No. 206 to License No. DPR-52
3. Amendment No. 163 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

9303290288

Tennessee Valley Authority
ATTN: Dr. Mark O. Medford

Browns Ferry Nuclear Plant

cc:
Mr. John B. Waters, Chairman
Tennessee Valley Authority
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400 West Summit Hill Drive
Knoxville, Tennessee 37902

Mr. J. R. Bynum, Vice President
Nuclear Operations
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Decatur, Alabama 35602

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
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 September 28, 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.

9303290302

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. 191, 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

Victor Nerses for

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: March 18, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 191

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
3.6/4.6-13	3.6/4.6-13
3.6/4.6-14	3.6/4.6-14
3.6/4.6-32	3.6/4.6-32*
3.6/4.6-33	3.6/4.6-33

LIMITING CONDITIONS FOR OPERATION

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.

    a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

    b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.

SURVEILLANCE REQUIREMENTS

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

4.6.G. Structural Integrity

3. For Unit 1 an augmented inservice surveillance program shall be performed to monitor potential corrosive effects of chloride residue released during the March 22, 1975 fire. The augmented inservice surveillance program is specified as follows:

   a. Browns Ferry Mechanical Maintenance Instruction 53, dated September 22, 1975, paragraph 4, defines the liquid penetrant examinations required during the first, second, third and fourth refueling outages following the fire restoration.

   b. Browns Ferry Mechanical Maintenance Instruction 46, dated July 18, 1975. Appendix B, defines the liquid penetrant examinations required during the sixth refueling outage following the fire restoration.

3.6.E/4.6.E (Cont'd)

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Steady-state operation without forced recirculation will not be permitted for more than 12 hours. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50% of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

3.6.G/4.6.G (Cont'd)

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in plant procedures to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

An augmented inservice surveillance program is required to determine whether any stress corrosion has occurred in any stainless steel piping, stainless components, and highly-stressed alloy steel such as hanger springs, as a result of environmental conditions associated with the March 22, 1975 fire.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)

2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code

3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)

4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

5. Mechanical Maintenance Instruction 46 (Mechanical Equipment, Concrete, and Structural Steel Cleaning Procedure for Residue From Plant Fire - Units 1 and 2)

6. Mechanical Maintenance Instruction 53 (Evaluation of Corrosion Damage of Piping Components Which Were Exposed to Residue From March 22, 1975 Fire)

7. Plant Safety Analysis (BFNP FSAR Subsection 4.12)

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. 206
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 September 28, 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 206 , 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

Victor Nerses for

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: March 18, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 206

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
3.6/4.6-13	3.6/4.6-13
3.6/4.6-14	3.6/4.6-14
3.6/4.6-32	3.6/4.6-32*
3.6/4.6-33	3.6/4.6-33

LIMITING CONDITIONS FOR OPERATION

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.

   a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a COLD SHUTDOWN CONDITION or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

SURVEILLANCE REQUIREMENTS

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.G Structural Integrity 3.6.G.1 (Cont'd) b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.	

3.6.E/4.6.E (Cont'd)

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Operation without forced recirculation is permitted for up to 12 hours when the reactor is not in the RUN mode. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring at least one recirculation pump to be operable while in the RUN mode (i.e., requiring a manual scram if both recirculation pumps are tripped) provides protection against the potential occurrence of core thermal-hydraulic instabilities at low flow conditions.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50% of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in plant procedures to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

An augmented inservice surveillance program is required to determine whether any stress corrosion has occurred in any stainless steel piping, stainless components, and highly-stressed alloy steel such as hanger springs, as a result of environmental conditions associated with the March 22, 1975 fire.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)

2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code

3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)

4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

5. Mechanical Maintenance Instruction 46 (Mechanical Equipment, Concrete, and Structural Steel Cleaning Procedure for Residue From Plant Fire - Units 1 and 2)

6. Mechanical Maintenance Instruction 53 (Evaluation of Corrosion Damage of Piping Components Which Were Exposed to Residue From March 22, 1975 Fire)

7. Plant Safety Analysis (BFNP FSAR Subsection 4.12)

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. 163
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 September 28, 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. 163, 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

Victor Nerses for

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: March 18, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 163

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
3.6/4.6-13	3.6/4.6-13
3.6/4.6-14	3.6/4.6-14
3.6/4.6-32	3.6/4.6-32*
3.6/4.6-33	3.6/4.6-33

LIMITING CONDITIONS FOR OPERATION

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.

   a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

   b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.

SURVEILLANCE REQUIREMENTS

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

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3.6.E/4.6.E (Cont'd)

area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Steady-state operation without forced recirculation will not be permitted for more than 12 hours. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50 percent of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

3.6.G/4.6.G (Cont'd)

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in plant procedures to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

References

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)

2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code

3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)

4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)

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

ENCLOSURE 4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 191 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 206 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 163 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

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

1.0 INTRODUCTION

By letter dated September 28, 1992, the Tennessee Valley Authority (TVA) submitted a request for changes to the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3 Technical Specifications (TS). The requested changes would relocate the list of additional inspections required to be performed on certain circumferential pipe welds from TS 4.6.G to plant procedures. Guidance on these proposed changes was provided to all holders of operating licenses or construction permits for nuclear power reactors by Generic Letter (GL) 91-08, "Removal of Component Lists From Technical Specification," dated May 6, 1991. By relocating the lists of pipe welds from TS to plant procedures, any subsequent changes to these particular component lists would be controlled pursuant to provisions of 10 CFR 50.59 and TS Section 6, "Administrative Controls."

2.0 EVALUATION

Tennessee Valley Authority proposed to remove the lists of certain circumferential pipe welds which required additional inspections, from Units 1, 2, and 3 TS Section 4.6.G.2. The contents of these lists will be incorporated into the applicable TVA program and procedures, which are subject to the administrative controls prescribed in Section 6.8, "Procedural/Instructions and Programs," of the BFN TS. As is required by GL 91-08, proposed TS Section 4.6.G.2 contains an appropriate description of the scope of the components to which the TS requirements apply.

The BASES for TS Section 3.6.G/4.6.G provides a more specific description of the type of pipe welds that are listed. Proposed changes to the BASES states that the list of circumferential pipe welds to be inspected is in plant procedures. Furthermore, at the request of NRC staff, TVA submitted additional information by letter dated February 22, 1993, regarding the technical basis for selecting and inspecting these welds.

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After reviewing TVA's amendment application, the staff concludes that the proposed TS changes for BFN, Units 1, 2, and 3, are primarily administrative in nature and conform with the guidance of GL 91-08 and are therefore acceptable.

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

These amendments change a requirement with respect to installation or use and surveillance of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 55592). Accordingly, these 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: C. Beardslee and T. Ross

Date: March 18, 1993