January 8, 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

See Proposed Change to Feel Apecs.

Dear Mr. Medford:

SUBJECT: ISSUANCE OF AMENDMENTS FOR THE BROWNS FERRY NUCLEAR PLANT (TAC NOS. M81056, M81057, AND M81058) (TS 293)

The Commission has issued the enclosed Amendment Nos.190, 205 and 162 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 an application from the Tennessee Valley Authority (TVA) dated July 19, 1991, and are also based upon additional information provided by TVA letter dated October 24, 1991. The staff has reviewed TVA's proposed pressure/temperature technical specification limits for plant operation during heatup, cooldown, criticality, and hydrotest for 12 effective full-power years and concluded they are acceptable.

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

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

Enclosures:

- Amendment No190 to License No. DPR-33
- 2. Amendment No 205 to License No. DPR-52
- 3. Amendment No.162 to License No. DPR-68
- 4. Safety Evaluation

cc w/enclosures: See next page

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Tennessee Valley Authority ATTN: Dr. Mark O. Medford

cc: Mr. John B. Waters, Chairman Tennessee Valley Authority ET 12A 400 West Summit Hill Drive Knoxville, Tennessee 37902

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Site Quality Manager Browns Ferry Nuclear Plant Tennessee Valley Authority P. O. Box 2000 Decatur, Alabama 35602

AMENDMENT NO.190 FOR BROWNS AMENDMENT NO.205 FOR BROWNS AMENDMENT NO.162 FOR BROWNS DATE: January 8, 1993	FERRY UNIT I - DOCKET NO. 50-259, FERRY UNIT 2 - DOCKET NO. 50-260 AND FERRY UNIT 3 - DOCKET NO. 50-296
<u>DISTRIBUTION</u> Docket File NRC PDR Local PDR	
BFN Reading File	
S. Varga	14-E-4
G. Lainas	14-H-3
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J. WILLIAMS	DII
P. Nellogg R. Wilson	
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D. Hagan	MNBR-3302
E. Jordan	MNBB-3302
G. Hill (4 per docket)	P1-130 (12)
Wanda Jones	MNBB-7103
J. Calvo	14-E-4
C. Fairbanks	
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GPA/PA	2-G-5
FINDD-9112 P. Wilson	
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cc: Plant Service List

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

ENCLOSURE 4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-33 AMENDMENT NO. 205 TO FACILITY OPERATING LICENSE NO. DPR-52 AMENDMENT NO. 162 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

NUCLEAR REGULA

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By letters dated July 19, 1991 and October 24, 1991, the Tennessee Valley Authority (TVA), the licensee, requested permission to revise the pressure/ temperature (P/T) limits of the Browns Ferry Nuclear Plant (BFN) Technical Specifications (TS) for Units 1, 2, and 3. More specifically, TVA proposed to revise the P/T limits of TS Section 3.6 in accordance with the guidance of Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," dated July 12, 1988. TVA's proposed P/T limits were developed using the methodology of Regulatory Guide (RG) 1.99, Revision 2, as recommended by GL 88-11. These P/T limits establish bounding conditions for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest for a cumulative neutron fluence corresponding to 12 effective full-power years (EFPY).

2.0 EVALUATION

In evaluating TVA's proposed P/T limits, the staff referred to the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and GL 88-11.

Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2. Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects

9301200345 930108 PDR ADUCK 05000259 P PDR of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). GL 88-11 requested that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.1 Browns Ferry Nuclear Plant, Unit 1

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the BFN, Unit 1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 12 EFPY was circumferential weld WF-154 between Shell Courses 1 and 2 with 0.31% copper (Cu), 0.50% nickel (Ni), and an initial RT_{ndt} of 20°F.

For the limiting beltline material, circumferential weld WF-154, the staff calculated the ART to be $87.3^{\circ}F$ at 1/4T (T = reactor vessel beltline thickness) and $61.6^{\circ}F$ for 3/4T at 12 EFPY. The staff used a neutron fluence of $2.86E17 \text{ n/cm}^2$ at 1/4T and $1.98E17 \text{ n/cm}^2$ at 3/4T. The ART was determined per Section 1 of RG 1.99, Rev. 2, because no surveillance capsules have been removed from the Unit 1 reactor vessel. Substituting the ART of $87.3^{\circ}F$ into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states "An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the pre-service system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.A.2 of Appendix G.

Section IV.A.1 of Appendix G requires that the predicted Charpy USE at end of life be no less than 50 ft-lb. Based on data from the licensee and using the method in RG 1.99, Rev. 2, the staff calculated that the beltline material with the lowest predicted end of life Charpy USE at the end of life at 1/4T is the circumferential weld metal WF-154 at 51.5 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

TVA has not removed any of the surveillance capsules from BFN, Unit 1. All surveillance capsules presently contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

2.2 Browns Ferry Nuclear Plant, Unit 2

The staff evaluated the effect of neutron irradiation embrittlement on each belt-line material in the BFN, Unit 2 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 12 EFPY was longitudinal weld in Shell Course 1 with 0.25% copper (Cu), 0.35% nickel (Ni), and an initial RT_{rdt} of 10°F.

For the limiting beltline material, circumferential weld WF-154, the staff calculated the ART to be 70.4°F at 1/4T (T = reactor vessel beltline thickness) and 48.3°F for 3/4T at 12 EFPY. The staff used a neutron fluence of 4.08E17 n/cm² at 1/4T and 2.83E17 n/cm² at 3/4T. The ART was determined per Section 1 of RG 1.99, Rev. 2, because no surveillance capsules have been removed from the Unit 2 reactor vessel. Substituting the ART of 70.4°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states "An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the pre-service system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.A.2 of Appendix G.

Section IV.A.1 of Appendix G requires that the predicted Charpy USE at end of life be no less than 50 ft-lb. Based on data from the licensee and using the method in RG 1.99, Rev. 2, the staff calculated that the beltline material

with the lowest predicted end of life Charpy USE at the end of life at 1/4T is plate C2467-2 at 68.1 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

TVA has not removed any surveillance capsules from BFN, Unit 2. All surveillance capsules presently contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

2.3 Browns Ferry Nuclear Plant, Unit 3

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the BFN, Unit 3 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 12 EFPY was longitudinal weld in Shell Course 1 with 0.25% copper (Cu), 0.35% nickel (Ni), and an initial RT_{rdt} of 10°F.

For the limiting beltline material, circumferential weld WF-154, the staff calculated the ART to be 69.2° F at 1/4T (T = reactor vessel beltline thickness) and 47.4° F for 3/4T at 12 EFPY. The staff used a neutron fluence of 3.94E17 n/cm² at 1/4T and 2.73E17 n/cm² at 3/4T. The ART was determined per Section 1 of RG 1.99, Rev. 2, because no surveillance capsules have been removed from the Unit 3 reactor vessel. Substituting the ART of 69.2° F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states "An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.A.2 of Appendix G.

Section IV.A.1 of Appendix G requires that the predicted Charpy USE at end of life be no less than 50 ft-lb. Based on data from the licensee and using the method in RG 1.99, Rev. 2, the staff calculated that the beltline material with the lowest predicted end of life Charpy USE at the end of life at 1/4T is plate C3222-2 at 68.5 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

TVA has not removed any surveillance capsules from BFN, Unit 3. All surveillance capsules presently contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

3.0 <u>STATE_CONSULTATION</u>

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of these amendments. 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. The NRC staff has determined that these 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 (56 FR 64662). 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 staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, hydrotest, and criticality are valid through 12 EFPY because the proposed limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed P/T limits also satisfy GL 88-11 because the method in RG 1.99, Rev. 2 was used to calculate the ART. Therefore, TVA's proposed P/T limits, and associated changes, are acceptable for the BFN, Units 1, 2, and 3 Technical Specifications.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: C. Fairbanks

Date: January 8, 1993



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. 190 License No. DPR-33

- 1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Tennessee Valley Authority (TVA), the licensee, dated July 19, 1991, including supplemental information from TVA letter dated October 24, 1991, 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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.190, 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

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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: January 8, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 190

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

vii	vii*
viii	viii
3.6/4.6-3	3.6/4.6-3
3.6/4.6-4	3.6/4.6-4*
3.6/4.6-24	3.6/4.6-24
3.6/4.6-25	3.6/4.6-25*
3.6/4.6-26	3.6/4.6-26
3.6/4.6-27	3.6/4.6-27
3.6/4.6-28	3.6/4.6-28**
3.6/4.6-29	3.6/4.6-29**

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4.2.G	Surveillance Requirements for Control Room Isolation Instrumentation	3.2/4.2-56
4.2.H	Minimum Test and Calibration Frequency for Flood Protection Instrumentation	3.2/4.2-57
4.2.J	Seismic Monitoring Instrument Surveillance Requirements	3.2/4.2-58
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BFN Unit 1 AMENDMENT NO. 1 8 9, 190

1

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4

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

3.6.A. <u>Thermal and Pressurization</u> <u>Limitations</u>

- The beltline region of 4. reactor vessel temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of Figure 3.6-1. The applicability of this curve to these tests is extended to nonnuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.
 - 5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 100°F, and must remain above 100°F while under full tension.

SURVEILLANCE REQUIREMENTS

- 4.6.A. <u>Thermal and Pressurization</u> <u>Limitations</u>
 - 4. DELETED

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

BFN Unit 1 Amendment 190

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

- 3.6.A <u>Thermal and Pressurization</u> <u>Limitations</u> (Cont'd)
 - 6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.
 - 7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

SURVEILLANCE REQUIREMENTS

- 4.6.A <u>Thermal and Pressurization</u> <u>Limitations</u> (Cont'd)
 - Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.
 - 7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

BFN Unit 1



Figure 3.6.-1 (BFN Unit 1)

Curve No. 1 Minimum temperature for pressure tests such as required by Section XI. Minimum temperature of 195 °F is required for test pressure of 1,100 psig.

Curve No. 2

Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3

Minimum temperature for core operation (criticality) Includes additional margin required for 10CFR50, Appendix G, Par. IV.A.3 which became effective July 26, 1983.

Notes

These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel bettline materials, in accordance with Reg. Guide 1.99 Rev. 2, to compensate for radiation embrittlement for 12 EFPY.

Figure 3.6-1

BFN Unit l 3.6/4.6-24

Amendment 190

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BFN Unit 1 3.6/4.6-25 Amendment 190

3.6.A/4.6.A Thermal and Pressurization Limitations

The reactor vessel has been analyzed for cyclic stresses caused by the temperature and pressure transients that arise from reactor trips, normal startup and shutdown, etc. The analysis assumed a maximum uniform heatup and cooldown rate of 100°F per hour for normal startup and shutdown and demonstrated that normal startup and shutdown cycles are within the required stress limits of Section III of the ASME Boiler and Pressure Vessel Code (1965 Edition including Summer 1966 addenda).

The operating limit curves for the reactor vessel (see Figure 3.6-1) were established in accordance with the requirements of lOCFR50 Appendix G and Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition. The curves are based on a large postulated surface flaw, with a depth one-quarter of the vessel thickness, the referenced toughness, RT_{NDT} , and the stress intensity factors for the reactor vessel components.

The fracture toughness of ferritic steels decreases with exposure to fast neutrons (E > 1 MeV) and therefore, initial values of RT_{NDT} have been adjusted to account for radiation embrittlement in the beltline region of the reactor vessel where neutron fluences are greater than 10^{17} n/cm². An adjusted reference temperature based on neutron fluence, copper content, nickel content, and initial RT_{NDT} for the controlling material was established using the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure-temperature limit curve, Figure 3.6-1, Curves 1, 2, and 3, includes a shift in RT_{NDT} caused by the fluence corresponding to 12 effective full power years (EFPY) of operation.

Radiation embrittlement of the reactor vessel materials will be monitored periodically during operation by removing and evaluating, irradiation flux wires and Charpy impact specimens contained in capsules installed near the inside wall of the reactor vessel in the core region. After the first refueling outage, flux wires were removed and evaluated. The data were used to verify calculated neutron fluence and to predict cumulative neutron fluence after 12 EFPY. Capsules that are withdrawn in the future will contain flux wires and Charpy impact specimens. Data derived from these specimens will be used as input to future radiation embrittlement evaluations that will account for neutron fluences above those corresponding to 12 EFPY.

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

TVA letter dated May 15, 1987, proposed to withdraw the first set of reactor surveillance specimens from each reactor vessel at the end of each unit's cycle which most closely approximates 8.0 EFPY of operation. The reasoning was the development of an integrated surveillance program related to estimated fluence obtained from reactor vessel specimens prior to 8.0 EFPY would be premature because it would be based only on extrapolations of limited dosimetry measurements taken from unit 1 during the first cycle of operation. Dosimetry measurements for 8.0 EFPY would be more credible than cycle 1 dosimetry data. NRC letter dated December 2, 1988, stated that BFN could withdraw the first reactor vessel specimen from each reactor vessel at the end of each unit's cycle of operation that most closely approximates 8.0 EFPY of operation. After withdrawal of each unit's first sample, the remaining specimens will be withdrawn every 6.0 EFPY thereafter.

As described in paragraph 4.2.5 of the Safety Analysis Report, detailed stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The requirements for full tension boltup of the reactor vessel closure are based on the NDT temperature plus 60°F. This is derived from the requirements of the ASME code to which the vessel was built. The NDT temperature of the closure flanges, adjacent head, and shell material is a maximum of 40°F and a maximum of 10°F for the stud material. Therefore, the minimum temperature for full tension boltup is 40°F plus 60°F for a

BFN Unit 1

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

total of 100°F. The partial boltup is restricted to the full loading of eight studs at 70°F, which is stud NDT temperature (10°F) plus 60°F. The neutron radiation fluence at the closure flanges is well below 10^{17} nvt \geq 1 Mev; therefore, radiation effects will be minor and will not influence this temperature.

3.6.B/4.6.B Coolant Chemistry

Materials in the primary system are primarily 304 stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is continuously measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel.

Zircaloy does not exhibit similar stress corrosion failures. However, there are some operating conditions under which the dissolved oxygen content of the reactor coolant water could be higher than .2-.3 ppm, such as reactor STARTUP and Hot Standby. During these periods, the most restrictive limits for conductivity and chlorides have been established. When steaming rates exceed 100,000 lb/hr, boiling deaerates the reactor water. This reduces dissolved oxygen concentration and assures minimal chloride-oxygen content, which together tend to induce stress corrosion cracking.

When conductivity is in its normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWRs, however, where no additives are used and where near neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include operation of the reactor cleanup system, reducing the input of impurities and placing the reactor in the Cold Shutdown condition. The major benefit of Cold Shutdown is to reduce the temperature dependent corrosion rates and provide time for the cleanup system to reestablish the purity of the reactor coolant.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its

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Unit 1

3.6/4.6-28

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

normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Daily sampling is performed when increased chloride concentrations are most probable. Reactor coolant sampling is increased to once per shift when the continuous conductivity monitor is unavailable.

The basis for the equilibrium coolant iodine activity limit is a computed dose to the thyroid of 36 rem at the exclusion distance during the two-hour period following a steam line break. This dose is computed with the conservative assumption of a release of 140,000 lbs of coolant prior to closure of the isolation valves, and a X/Q value of $3.4 \times 10^{-4} \text{ Sec/m}^3$.

The maximum activity limit during a short term transient is established from consideration of a maximum iodine inhalation dose less than 300 rem. The probability of a steam line break accident coincident with an iodine concentration transient is significantly lower than that of the accident alone, since operation of the reactor with iodine levels above the equilibrium value is limited to 5 percent of total operation.

The sampling frequencies are established in order to detect the occurrence of an iodine transient which may exceed the equilibrium concentration limit, and to assure that the maximum coolant iodine concentrations are not exceeded. Additional sampling is required following power changes and off-gas transients, since present data indicate that the iodine peaking phenomenon is related to these events.

3.6.C/4.6.C <u>Coolant Leakage</u>

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite ac power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of five gpm, as specified in 3.6.C, the experimental and analytical data

3.6/4.6-29

AMENDMENT NO. 170

BFN Unit 1



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. 205 License No. DPR-52

- 1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Tennessee Valley Authority (TVA), the licensee, dated July 19, 1991, including supplemental information from TVA letter dated October 24, 1991, 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. 205, 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

Voredende G. Hebd

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: January 8, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 205

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.

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REMOVE	INSERT
vii	vii* vii
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3.6/4.6-4 3.6/4.6-24	3.6/4.6-24
3.6/4.6-25	3.6/4.6-25* 3.6/4.6-26
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3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

3.6.A. <u>Thermal and Pressurization</u> Limitations (Cont'd)

- 4. The beltline region of reactor vessel temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of Figure 3.6-1. The applicability of this curve to these tests is extended to nonnuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.
- 5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 100°F, and must remain above 100°F while under full tension.

SURVEILLANCE REQUIREMENTS

- 4.6.A. <u>Thermal and Pressurization</u> <u>Limitations (Cont'd)</u>
 - 4. DELETED

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

BFN Unit 2 Amendment 205

3.6/4.6 PRIMARY S_TEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

- 3.6.A <u>Thermal and Pressurization</u> <u>Limitations</u> (Cont'd)
 - 6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.
 - 7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.

SURVEILLANCE REQUIREMENTS

- 4.6.A <u>Thermal and Pressurization</u> <u>Limitations</u> (Cont'd)
 - 6. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.
 - 7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.



Figure 3.6.-1 (BFN Units 2 and 3)

Curve No. 1

Minimum temperature for pressure tests such as required by Section XI. Minimum temperature of 185 °F is required for test pressure of 1,100 psig.

Curve No. 2

Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3

Minimum temperature for core operation (criticality) includes additional margin required for 10CFR50, Appendix G, Par. IV.A.3 which became effective July 26, 1983.

Notes

These curves Include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel bettine materials, in accordance with Reg. Guide 1.99 Rev. 2, to compensate for radiation embrittlement for 12 EFPY.



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BFN Unit 2 3.6/4.6-25 Amendment 205

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3.6.A/4.6.A Thermal and Pressurization Limitations

The reactor vessel has been analyzed for cyclic stresses caused by the temperature and pressure transients that arise from reactor trips, normal startup and shutdown, etc. The analysis assumed a maximum uniform heatup and cooldown rate of 100°F per hour for normal startup and shutdown and demonstrated that normal startup and shutdown cycles are within the required stress limits of Section III of the ASME Boiler and Pressure Vessel Code (1965 Edition including Summer 1966 addenda).

The operating limit curves for the reactor vessel (see Figure 3.6-1) were established in accordance with the requirements of 10CFR50 Appendix G and Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition. The curves are based on a large postulated surface flaw, with a depth one-quarter of the vessel thickness, the referenced toughness, RT_{NDT} , and the stress intensity factors for the reactor vessel components.

The fracture toughness of ferritic steels decreases with exposure to fast neutrons (E > 1 MeV) and therefore, initial values of RT_{NDT} have been adjusted to account for radiation embrittlement in the beltline region of the reactor vessel where neutron fluences are greater than 10^{17} n/cm². An adjusted reference temperature based on neutron fluence, copper content, nickel content, and initial RT_{NDT} for the controlling material was established using the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure-temperature limit curve, Figure 3.6-1, Curves 1, 2, and 3, includes a shift in RT_{NDT} caused by the fluence corresponding to 12 effective full power years (EFPY) of operation.

Radiation embrittlement of the reactor vessel materials will be monitored periodically during operation by removing and evaluating, irradiation flux wires and Charpy impact specimens contained in capsules installed near the inside wall of the reactor vessel in the core region. After the first refueling outage, flux wires were removed and evaluated. The data were used to verify calculated neutron fluence and to predict cumulative neutron fluence after 12 EFPY. Capsules that are withdrawn in the future will contain flux wires and Charpy impact specimens. Data derived from these specimens will be used as input to future radiation embrittlement evaluations that will account for neutron fluences above those corresponding to 12 EFPY.

BFN Unit 2 **3.6/4.6–26** Amendment 205

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

TVA letter dated May 15, 1987, proposed to withdraw the first set of reactor surveillance specimens from each reactor vessel at the end of each unit's cycle which most closely approximates 8.0 EFPY of operation. The reasoning was the development of an integrated surveillance program related to estimated fluence at this time would be premature because it would be based only on extrapolations of limited dosimetry measurements taken from unit 1 during the first cycle. Dosimetry measurements for 8.0 EFPY would be more credible than cycle 1 dosimetry data. NRC letter dated December 2, 1988, agreed and stated that BFN could withdraw the first specimen from each reactor vessel at the end of each unit's cycle of operation most closely approximates 8.0 EFPY of operation. After withdrawal of each unit's first sample, the remaining specimens will be withdrawn every 6.0 EFPY thereafter.

As described in paragraph 4.2.5 of the Safety Analysis Report, detailed stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The requirements for full tension boltup of the reactor vessel closure are based on the NDT temperature plus 60°F. This is derived from the requirements of the ASME code to which the vessel was built. The NDT temperature of the closure flanges, adjacent head, and shell material is a maximum of 40°F and a maximum of 10°F for the stud material. Therefore, the minimum temperature for full tension boltup is 40°F plus 60°F for a total of 100°F. The partial boltup is restricted to the full loading of

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

eight studs at 70°F, which is stud NDT temperature (10°F) plus 60°F. The neutron radiation fluence at the closure flanges is well below 10^{17} nvt \geq 1 Mev; therefore, radiation effects will be minor and will not influence this temperature.

3.6.B/4.6.B <u>Coolant Chemistry</u>

Materials in the primary system are primarily 304 stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is continuously measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel.

Zircaloy does not exhibit similar stress corrosion failures. However, there are some operating conditions under which the dissolved oxygen content of the reactor coolant water could be higher than .2-.3 ppm, such as reactor startup and hot standby. During these periods, the most restrictive limits for conductivity and chlorides have been established. When steaming rates exceed 100,000 lb/hr, boiling deaerates the reactor water. This reduces dissolved oxygen concentration and assures minimal chloride-oxygen content, which together tend to induce stress corrosion cracking.

When conductivity is in its normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors. conductivities are in fact high due to purposeful addition of additives. In the case of BWRs, however, where no additives are used and where near neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include operation of the reactor cleanup system, reducing the input of impurities and placing the reactor in the Cold Shutdown condition. The major benefit of Cold Shutdown is to reduce the temperature dependent corrosion rates and provide time for the cleanup system to reestablish the purity of the reactor coolant.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within

BFN Unit 2 AMENDMENT NO. 170

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

its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Daily sampling is performed when increased chloride concentrations are most probable. Reactor coolant sampling is increased to once per shift when the continuous conductivity monitor is unavailable.

The basis for the equilibrium coolant iodine activity limit is a computed dose to the thyroid of 36 rem at the exclusion distance during the two-hour period following a steam line break. This dose is computed with the conservative assumption of a release of 140,000 lbs of coolant prior to closure of the isolation valves, and a X/Q value of 3.4×10^{-4} Sec/m³.

The maximum activity limit during a short term transient is established from consideration of a maximum iodine inhalation dose less than 300 rem. The probability of a steam line break accident coincident with an iodine concentration transient is significantly lower than that of the accident alone, since operation of the reactor with iodine levels above the equilibrium value is limited to 5 percent of total operation.

The sampling frequencies are established in order to detect the occurrence of an iodine transient which may exceed the equilibrium concentration limit, and to assure that the maximum coolant iodine concentrations are not exceeded. Additional sampling is required following power changes and off-gas transients, since present data indicate that the iodine peaking phenomenon is related to these events.

3.6.C/4.6.C <u>Coolant Leakage</u>

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite ac power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of

BFN Unit 2 AMENDMENT NO. 170



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. 162 License No. DPR-68

- 1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Tennessee Valley Authority (TVA), the licensee, dated July 19, 1991, including supplemental information from TVA letter dated October 24, 1991, 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 in 10 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. 162, 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 Q. Hebds

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: January 8, 1993

- 2 -

ATTACHMENT TO LICENSE AMENDMENT NO. 162

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

vii	vii*
viii	viii
3.6/4.6-3	3.6/4.6-3
3.6/4.6-4	3.6/4.6-4*
3.6/4.6-24	3.6/4.6-24
3.6/4.6-25	3.6/4.6-25*
3.6/4.6-26	3.6/4.6-26
3.6/4.6-27	3.6/4.6-27
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3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

3.6.A. <u>Thermal and Pressurization</u> <u>Limitations</u>

- The beltline region of 4. reactor vessel temperatures during inservice hydrostatic or leak testing shall be at or above the temperatures shown on curve #1 of Figure 3.6-1. The applicability of this curve to these tests is extended to nonnuclear heatup and ambient loss cooldown associated with these tests only if the heatup and cooldown rates do not exceed 15°F per hour.
- 5. The reactor vessel head bolting studs may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are above 70°F. Before loading the flanges any more, the vessel flange and head flange must be greater than 100°F, and must remain above 100°F while under full tension.

SURVEILLANCE REQUIREMENTS

- 4.6.A. <u>Thermal and Pressurization</u> <u>Limitations</u>
 - 4. DELETED

5. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

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3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

3.6.A <u>Thermal and Pressurization</u> <u>Limitations</u> (Cont'd)

- 6. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.
- 7. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and bottom head drain are within 145°F.

SURVEILLANCE REQUIREMENTS

- 4.6.A <u>Thermal and Pressurization</u> <u>Limitations</u> (Cont'd)
 - 6. Prior to and during STARTUP of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.
 - 7. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

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Figure 3.6.-1 (BFN Units 2 and 3)

Curve No. 1

Minimum temperature for pressure tests such as required by Section XI. Minimum temperature of 185 °F is required for test pressure of 1,100 psig.

Curve No. 2

Minimum temperature for mechanical heatup or cooldown following nuclear shutdown.

Curve No. 3

Minimum temperature for core operation (criticality) includes additional margin required for 10CFR50, Appendix G, Par. IV.A.3 which became effective July 26, 1983.

Notes

These curves include sufficient margin to provide protection against feedwater nozzle degradation. The curves allow for shifts in RT_{NDT} of the Reactor vessel bettline materials, In accordance with Reg. Guide 1.99 Rev. 2, to compensate for radiation embrittlement for 12 EFPY.

Figure 3.6-1

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3.6.A/4.6.A Thermal and Pressurization Limitations

The reactor vessel has been analyzed for cyclic stresses caused by the temperature and pressure transients that arise from reactor trips, normal startup and shutdown, etc. The analysis assumed a maximum uniform heatup and cooldown rate of 100°F per hour for normal startup and shutdown and demonstrated that normal startup and shutdown cycles are within the required stress limits of Section III of the ASME Boiler and Pressure Vessel Code (1965 Edition including Summer 1966 addenda).

The operating limit curves for the reactor vessel (see Figure 3.6-1) were established in accordance with the requirements of 10CFR50 Appendix G and Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition. The curves are based on a large postulated surface flaw, with a depth one-quarter of the vessel thickness, the referenced toughness, RT_{NDT} , and the stress intensity factors for the reactor vessel components.

The fracture toughness of ferritic steels decreases with exposure to fast neutrons (E > 1 MeV) and therefore, initial values of RT_{NDT} have been adjusted to account for radiation embrittlement in the beltline region of the reactor vessel where neutron fluences are greater than 10^{17} n/cm². An adjusted reference temperature based on neutron fluence, copper content, nickel content, and initial RT_{NDT} for the controlling material was established using the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure-temperature limit curve, Figure 3.6-1, Curves 1, 2, and 3, includes a shift in RT_{NDT} caused by the fluence corresponding to 12 effective full power years (EFPY) of operation.

Radiation embrittlement of the reactor vessel materials will be monitored periodically during operation by removing and evaluating, irradiation flux wires and Charpy impact specimens contained in capsules installed near the inside wall of the reactor vessel in the core region. After the first refueling outage, flux wires were removed and evaluated. The data were used to verify calculated neutron fluence and to predict cumulative neutron fluence after 12 EFPY. Capsules that are withdrawn in the future will contain flux wires and Charpy impact specimens. Data derived from these specimens will be used as input to future radiation embrittlement evaluations that will account for neutron fluences above those corresponding to 12 EFPY.

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

TVA letter dated May 15, 1987, proposed to withdraw the first set of reactor surveillance specimens from each reactor vessel at the end of each unit's cycle which most closely approximates 8.0 EFPY of operation. The reasoning was the development of an integrated surveillance program related to estimated fluence at this time would be premature because it would be based only on extrapolations of limited dosimetry measurements taken from unit 1 during the first cycle. Dosimetry measurements for 8.0 EFPY would be more credible than cycle 1 dosimetry data. NRC letter dated December 2, 1988, agreed and stated that BFN could withdraw the first specimen from each reactor vessel at the end of each unit's cycle of operation most closely approximates 8.0 EFPY of operation. After withdrawal of each unit's first sample, the remaining specimens will be withdrawn every 6.0 EFPY thereafter.

As described in paragraph 4.2.5 of the Safety Analysis Report, detailed stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The requirements for full tension boltup of the reactor vessel closure are based on the NDT temperature plus 60°F. This is derived from the requirements of the ASME code to which the vessel was built. The NDT temperature of the closure flanges, adjacent head, and shell material is a maximum of 40°F and a maximum of 10°F for the stud material. Therefore, the minimum temperature for full tension boltup is 40°F plus 60°F for a total of 100°F. The partial boltup is restricted to the full loading of

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

eight studs at 70°F, which is stud NDT temperature (10°F) plus 60°F. The neutron radiation fluence at the closure flanges is well below 10^{17} nvt \geq 1 Mev; therefore, radiation effects will be minor and will not influence this temperature.

3.6.B/4.6.B <u>Coolant Chemistry</u>

Materials in the primary system are primarily 304 stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is continuously measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel.

Zircaloy does not exhibit similar stress corrosion failures. However, there are some operating conditions under which the dissolved oxygen content of the reactor coolant water could be higher than .2-.3 ppm, such as reactor startup and hot standby. During these periods, the most restrictive limits for conductivity and chlorides have been established. When steaming rates exceed 100,000 lb/hr, boiling deaerates the reactor water. This reduces dissolved oxygen concentration and assures minimal chloride-oxygen content, which together tend to induce stress corrosion cracking.

When conductivity is in its normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWRs, however, where no additives are used and where near neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include operation of the reactor cleanup system, reducing the input of impurities and placing the reactor in the Cold Shutdown condition. The major benefit of Cold Shutdown is to reduce the temperature dependent corrosion rates and provide time for the cleanup system to reestablish the purity of the reactor coolant.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within

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

its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Daily sampling is performed when increased chloride concentrations are most probable. Reactor coolant sampling is increased to once per shift when the continuous conductivity monitor is unavailable.

The basis for the equilibrium coolant iodine activity limit is a computed dose to the thyroid of 30 rem at the exclusion distance during the two-hour period following a steam line break. This dose is computed with the conservative assumption of a release of 140,000 lbs of coolant prior to closure of the isolation valves, and a X/Q value of $2.9 \times 10^{-4} \text{ Sec/m}^3$.

The maximum activity limit during a short term transient is established from consideration of a maximum iodine inhalation dose less than 300 rem. The probability of a steam line break accident coincident with an iodine concentration transient is significantly lower than that of the accident alone, since operation of the reactor with iodine levels above the equilibrium value is limited to 5 percent of total operation.

The sampling frequencies are established in order to detect the occurrence of an iodine transient which may exceed the equilibrium concentration limit, and to assure that the maximum coolant iodine concentrations are not exceeded. Additional sampling is required following power changes and off-gas transients, since present data indicate that the iodine peaking phenomenon is related to these events.

3.6.C/4.6.C <u>Coolant Leakage</u>

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite ac power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USABC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of five gpm, as specified in 3.6.C, the experimental and analytical data

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