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.

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.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

Frederick J. Hebdon

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**

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

LIMITING CONDITIONS FOR OPERATION

3.6.A. Thermal and Pressurization Limitations

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. Thermal and Pressurization Limitations

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.

LIMITING CONDITIONS FOR OPERATION

3.6.A Thermal and Pressurization Limitations (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 Thermal and Pressurization Limitations (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.

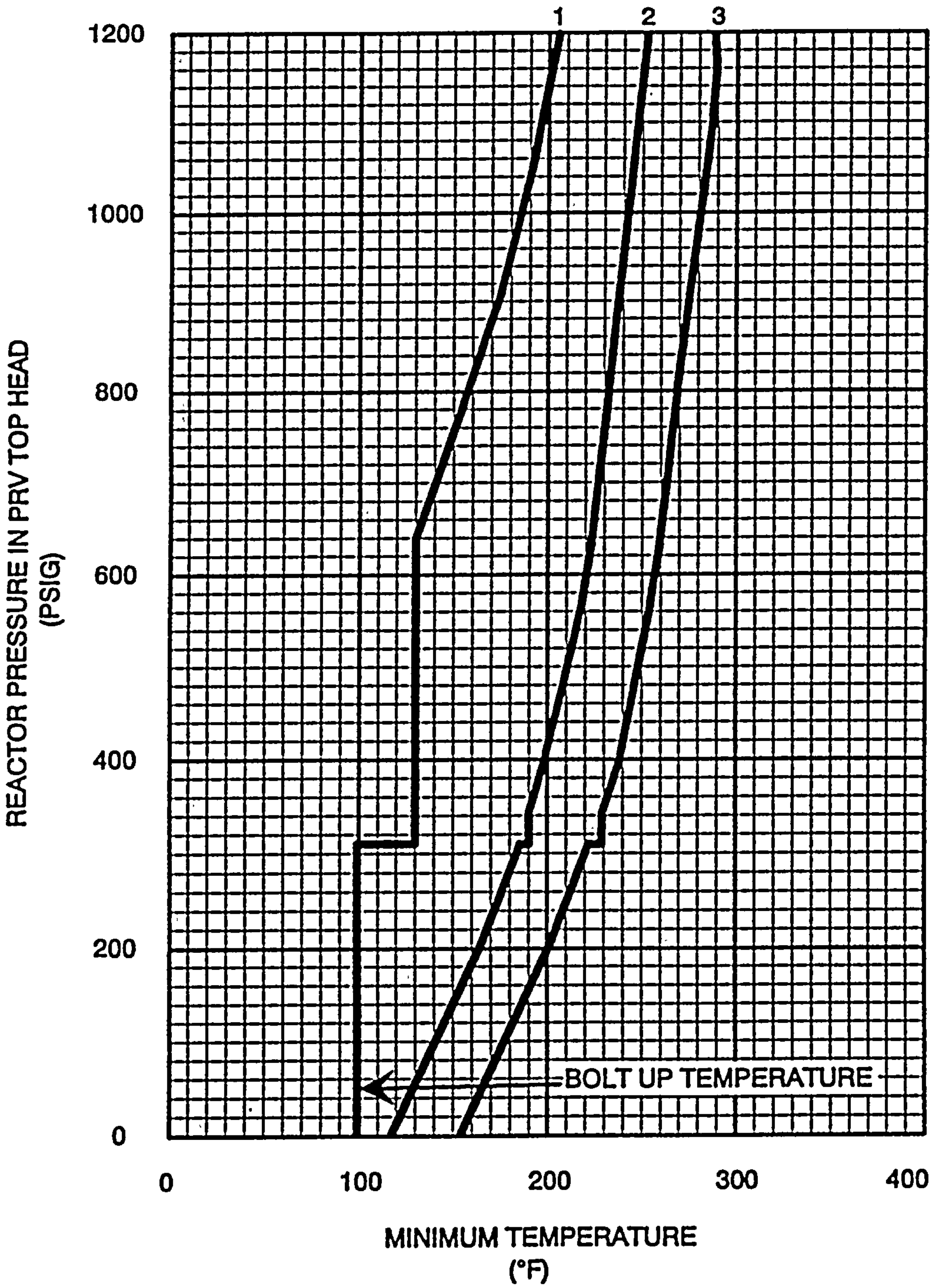

Figure 3.6.-1
(BFN Unit 1)

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 beltline 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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3.6/4.6 BASES

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/cm^2. 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.

3.6/4.6 BASES

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

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

3.6/4.6 BASES

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 3.4×10^{-4} 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 Coolant Leakage

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

UNITED STATES
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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) Technical Specifications

   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

Frederick J. Hebdon

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.

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**

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.A. Thermal and Pressurization Limitations (Cont'd)	4.6.A. Thermal and Pressurization 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.	4. DELETED
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.	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.

LIMITING CONDITIONS FOR OPERATION

3.6.A Thermal and Pressurization Limitations (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 Thermal and Pressurization Limitations (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.

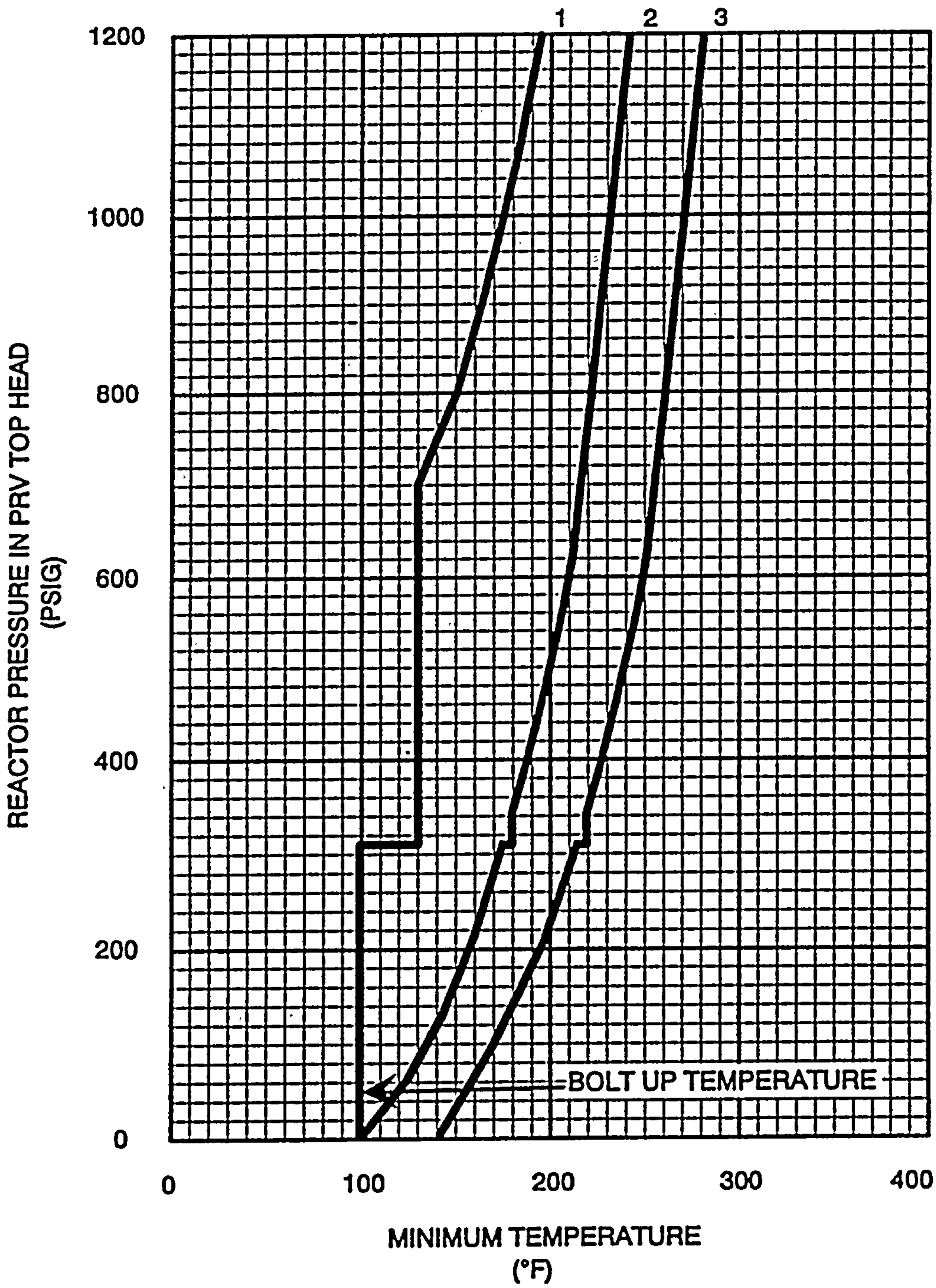

Figure 3.6.-1
(BFN Units 2 and 3)

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 beltline 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/cm^2. 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

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 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

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/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 Coolant Leakage

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

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 J. Hebdon

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. 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
3.6/4.6-28	3.6/4.6-28**
3.6/4.6-29	3.6/4.6-29**

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

3.6.A. Thermal and Pressurization Limitations

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. Thermal and Pressurization Limitations

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.

LIMITING CONDITIONS FOR OPERATION

3.6.A Thermal and Pressurization Limitations (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 Thermal and Pressurization Limitations (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.

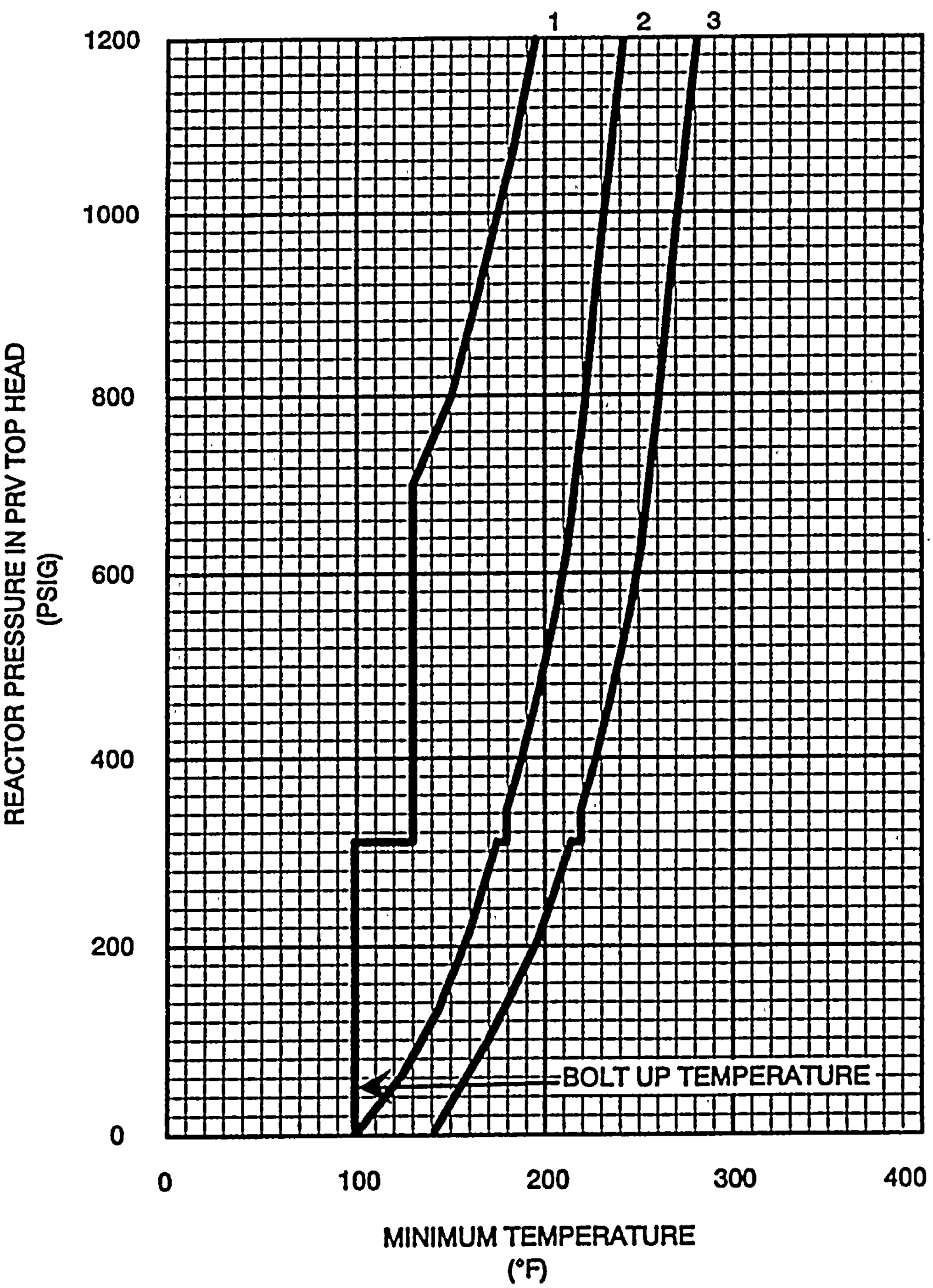

Figure 3.6-1

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 beltline materials, in accordance with Reg. Guide 1.99 Rev. 2, to compensate for radiation embrittlement for 12 EFPY.

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3.6/4.6 BASES

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/cm^2. 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

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 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

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 2.9×10^{-4} 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 Coolant Leakage

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