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August 3, 1989

POSTED
Amdt 170 to
DPR-52

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

Mr. Oliver D. Kingsley, Jr.
Senior Vice President, Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley:

SUBJECT: REVISION TO TECHNICAL SPECIFICATIONS PERTAINING TO SURVEILLLANCE REQUIREMENT 4.6.A.3 AND BASES SECTION 3.6/4.6 - (TAC 73141, 73142 73143) (TS 270) - BROWNS FERRY NUCLEAR PLANTS, UNITS 1, 2, AND 3

The Commission has issued the enclosed Amendment Nos. 170, 170, and 141to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments are in response to your application dated May 15, 1989. This amendment corrects Technical Specification Surveillance Requirement 4.6.A.3 to comply with 10 CFR 50, Appendix H and revises the Bases section to reflect the specimen withdrawal program agreed upon by TVA and the NRC.

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

Sincerely,

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 170 to License No. DPR-33
2. Amendment No. 170to License No. DPR-52
3. Amendment No. 141to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

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cc:
General Counsel
Tennessee Valley Authority
400 West Summit Hill Drive
E11 B33
Knoxville, Tennessee 37902

Mr. F. L. Moreadith
Vice President, Nuclear Engineering
Tennessee Valley Authority
400 West Summit Hill Drive
W12 A12
Knoxville, Tennessee 37902

Vice President and Nuclear
Technical Director
Tennessee Valley Authority
5N 157B Lookout Place
Chattanooga, Tennessee 37402-2801

Mr. M. J. Ray, Acting Director
Nuclear Safety and Licensing
Tennessee Valley Authority
5N 157B Lookout Place
Chattanooga, Tennessee 37402-2801

Mr. O. J. Zeringue
Site Director
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P. O. Box 2000
Decatur, Alabama 35602

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

Mr. G. Campbell
Plant Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P. O. Box 2000
Decatur, Alabama 35602

Chairman, Limestone County Commission
P. O. Box 188
Athens, Alabama 35611

Claude Earl Fox, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36130

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Atlanta, Georgia 30323

Mr. Danny Carpenter
Senior Resident Inspector
Browns Ferry Nuclear Plant
U.S. Nuclear Regulatory Commission
Route 12, Box 637
Athens, Alabama 35611

Dr. Henry Myers, Science Advisor
Committee on Interior
and Insular Affairs
U.S. House of Representatives
Washington, D.C. 20515

Tennessee Valley Authority
Rockville Office
11921 Rockville Pike
Suite 402
Rockville, Maryland 20852

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. 170
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 15, 1989, omplies 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 170 , 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 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 3, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 170

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 pages* are provided to maintain document completeness.

REMOVE	INSERT
3.6/4.6-1	3.6/4.6-1*
3.6/4.6-2	3.6/4.6-2
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
3.6/4.6-30	3.6/4.6-30
3.6/4.6-31	3.6/4.6-31*

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To assure the integrity and safe operation of the reactor coolant system.

Specification

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the following parameters shall be recorded and reactor coolant temperature determined at 15-minute intervals until 3 successive readings at each given location are within 5°F.

   a. Steam Dome Pressure (Convert to upper vessel region temperature)

   b. Reactor bottom drain temperature

   c. Recirculation loops A and B

   d. Reactor vessel bottom head temperature

   e. Reactor vessel shell adjacent to shell flange

LIMITING CONDITIONS FOR OPERATION

3.6.A. Thermal and Pressurization Limitations

2. During all operations with a critical core, other than for low-level physics tests, except when the vessel is vented, the reactor vessel shell and fluid temperatures shall be at or above the temperature of curve #3 of Figure 3.6-1.

3. During heatup by nonnuclear means, except when the vessel is vented or as indicated in 3.6.A.4, during cooldown following nuclear shutdown, or during low-level physics tests, the reactor vessel temperature shall be at or above the temperatures of curve #2 of Figure 3.6-1 until removing tension on the head stud bolts as specified in 3.6.A.5.

SURVEILLANCE REQUIREMENTS

4.6.A. Thermal and Pressurization Limitations

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every 15 minutes during inservice hydrostatic or leak testing when the vessel pressure is > 312 psig.

3. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-82.

3.6/4.6 BASES

3.6.A/4.6.A Thermal and Pressurization Limitations

The vessel has been analyzed for stresses caused by thermal and pressure transients. Heating and cooling transients throughout plant life at uniform rates of 100°F per hour were considered in the temperature range of 100 to 546°F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code (65 Edition including Summer 1966 addenda).

Operating limits on the reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, as a guide. These operating limits assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessel shell remote from discontinuities. For the purpose of setting these operating limits the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (65 Edition to Summer 1966 addenda).

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the RPV. Two types of information are needed in this analysis: 1) A relationship between the change in fracture toughness of the RPV steel and the neutron fluence (integrated neutron flux), and 2) a measure of the neutron fluence at the point of interest in the RPV wall.

A relationship between neutron fluence and change in Charpy V, 30-foot pound transition temperature has been developed for SA302B/SA533 steel based on at least 35 experimental data points as shown in Figure 3.6-2. In turn, this change in transition temperature can be related to a change in the temperature ordinate shown in Figure G 2110-1 in Appendix G of Section III of the Boiler Code.

The neutron fluence at any point in the pressure vessel wall can be computed from core physics data. The neutron fluence can also be measured experimentally on the ID of the vessel wall. At present valid experimental measurements can be made only over time periods of less than five years because of the limitations of the dosimeter materials. This causes no problem because of the exact relationship between thermal power produced and the number of neutrons produced from a given core geometry. A single experimental measurement in a time period of one year can be used to predict the fluence for the life of the plant in terms of thermal power output if no great changes in core geometry are made.

3.6/4.6 <u>BASES</u>

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

The vessel pressurization temperatures at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence and from Figure 3.6-2. For heatup or cooldown and core operation, see curves Nos. 2 and 3 on Figure 3.6-1. During the first fuel cycle, only calculated neutron fluence values can be used. At the first refueling, neutron dosimeter wires which are installed adjacent to the vessel wall can be removed to verify the calculated neutron fluence. As more experience is gained in calculating the fluence the need to verify it experimentally will disappear. Because of the many experimental points used to derive Figure 3.6-2, there is no need to reverify if for technical reasons, but in case verification is required for other reasons, three sets of mechanical test specimens representing the base metal, weld metal and weld heat affected zone metal have been placed in the vessel. These can be removed and tested as required.

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 ≥ 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.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

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

suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

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

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

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

REFERENCE

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

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 84.1 percent of nuclear boiler rated steam flow at a reference pressure of (1,105 + 1 percent) psig. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves operable, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their setpoints are within the ± 1 percent tolerance. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.

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

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

<u>REFERENCES</u>

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)
2. Amendment 22 in response to AEC Question 4.2 of December 6, 1971.
3. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)
4. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilliland to F. E. Kruesi, August 29, 1973
5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda

3.6.E/4.6.E <u>Jet Pumps</u>

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system

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. 170
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 15, 1989, 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. 170, 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 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 3, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 170

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 pages* are provided to maintain document completeness.

REMOVE	INSERT
3.6/4.6-1	3.6/4.6-1*
3.6/4.6-2	3.6/4.6-2
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
3.6/4.6-30	3.6/4.6-30
3.6/4.6-31	3.6/4.6-31*

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To assure the integrity and safe operation of the reactor coolant system.

Specification

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the following parameters shall be recorded and reactor coolant temperature determined at 15-minute intervals until 3 successive readings at each given location are within 5°F.

   a. Steam Dome Pressure (Convert to upper vessel region temperature)

   b. Reactor bottom drain temperature

   c. Recirculation loops A and B

   d. Reactor vessel bottom head temperature

   e. Reactor vessel shell adjacent to shell flange

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

3.6.A. Thermal and Pressurization Limitations

2. During all operations with a critical core, other than for low-level physics tests, except when the vessel is vented, the reactor vessel shell and fluid temperatures shall be at or above the temperature of curve #3 of Figure 3.6-1.

3. During heatup by nonnuclear means, except when the vessel is vented or as indicated in 3.6.A.4, during cooldown following nuclear shutdown, or during low-level physics tests, the reactor vessel temperature shall be at or above the temperatures of curve #2 of Figure 3.6-1 until removing tension on the head stud bolts as specified in 3.6.A.5.

SURVEILLANCE REQUIREMENTS

4.6.A. Thermal and Pressurization Limitations

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every 15 minutes during inservice hydrostatic or leak testing when the vessel pressure is > 312 psig.

3. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-82.

3.6/4.6 BASES

3.6.A/4.6.A Thermal and Pressurization Limitations

The vessel has been analyzed for stresses caused by thermal and pressure transients. Heating and cooling transients throughout plant life at uniform rates of 100°F per hour were considered in the temperature range of 100 to 546°F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code (65 Edition including Summer 1966 addenda).

Operating limits on the reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, as a guide. These operating limits assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessel shell remote from discontinuities. For the purpose of setting these operating limits the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (65 Edition to Summer 1966 addenda).

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the RPV. Two types of information are needed in this analysis: 1) A relationship between the change in fracture toughness of the RPV steel and the neutron fluence (integrated neutron flux), and 2) a measure of the neutron fluence at the point of interest in the RPV wall.

A relationship between neutron fluence and change in Charpy V, 30-foot pound transition temperature has been developed for SA302B/SA533 steel based on at least 35 experimental data points as shown in Figure 3.6-2. In turn, this change in transition temperature can be related to a change in the temperature ordinate shown in Figure G 2110-1 in Appendix G of Section III of the Boiler Code.

The neutron fluence at any point in the pressure vessel wall can be computed from core physics data. The neutron fluence can also be measured experimentally on the ID of the vessel wall. At present valid experimental measurements can be made only over time periods of less than five years because of the limitations of the dosimeter materials. This causes no problem because of the exact relationship between thermal power produced and the number of neutrons produced from a given core geometry. A single experimental measurement in a time period of one year can be used to predict the fluence for the life of the plant in terms of thermal power output if no great changes in core geometry are made.

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

The vessel pressurization temperatures at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence and from Figure 3.6-2. For heatup or cooldown and core operation, see curves Nos. 2 and 3 on Figure 3.6-1. During the first fuel cycle, only calculated neutron fluence values can be used. At the first refueling, neutron dosimeter wires which are installed adjacent to the vessel wall can be removed to verify the calculated neutron fluence. As more experience is gained in calculating the fluence the need to verify it experimentally will disappear. Because of the many experimental points used to derive Figure 3.6-2, there is no need to reverify if for technical reasons, but in case verification is required for other reasons, three sets of mechanical test specimens representing the base metal, weld metal and weld heat affected zone metal have been placed in the vessel. These can be removed and tested as required.

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

3.6/4.6 BASES

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

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

five gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

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

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

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

REFERENCE

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

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 84.1 percent of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves OPERABLE, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their setpoints are within the ± 1 percent tolerance. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.

3.6/4.6 BASES

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

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)
2. Amendment 22 in response to AEC Question 4.2 of December 6, 1971.
3. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)
4. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilleland to F. E. Kruesi, August 29, 1973
5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system

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. 141
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 15, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-68 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 141 , 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 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 3, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 141

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 pages* are provided to maintain document completeness.

REMOVE	INSERT
3.6/4.6-1	3.6/4.6-1*
3.6/4.6-2	3.6/4.6-2
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
3.6/4.6-30	3.6/4.6-30
3.6/4.6-31	3.6/4.6-31
3.6/4.6-32	3.6/4.6-32
3.6/4.6-33	3.6/4.6-33*

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To assure the integrity and safe operation of the reactor coolant system.

Specification

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

A. Thermal and Pressurization Limitations

1. During heatups and cooldowns, the following parameters shall be recorded and reactor coolant temperature determined at 15-minute intervals until 3 successive readings at each given location are within 5°F.

   a. Steam Dome Pressure (Convert to upper vessel region temperature)

   b. Reactor bottom drain temperature

   c. Recirculation loops A and B

   d. Reactor vessel bottom head temperature

   e. Reactor vessel shell adjacent to shell flange

LIMITING CONDITIONS FOR OPERATION

3.6.A. Thermal and Pressurization Limitations

2. During all operations with a critical core, other than for low-level physics tests, except when the vessel is vented, the reactor vessel shell and fluid temperatures shall be at or above the temperature of curve #3 of Figure 3.6-1.

3. During heatup by nonnuclear means, except when the vessel is vented or as indicated in 3.6.A.4, during cooldown following nuclear shutdown, or during low-level physics tests, the reactor vessel temperature shall be at or above the temperatures of curve #2 of Figure 3.6-1 until removing tension on the head stud bolts as specified in 3.6.A.5.

SURVEILLANCE REQUIREMENTS

4.6.A. Thermal and Pressurization Limitations

2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every 15 minutes during inservice hydrostatic or leak testing when the vessel pressure is > 312 psig.

3. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The number and type of specimens will be in accordance with GE report NEDO-10115. The specimens shall meet the intent of ASTM E 185-82.

3.6/4.6 BASES

3.6.A/4.6.A Thermal and Pressurization Limitations

The vessel has been analyzed for stresses caused by thermal and pressure transients. Heating and cooling transients throughout plant life at uniform rates of 100°F per hour were considered in the temperature range of 100 to 546°F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code (65 Edition including Summer 1966 Addenda).

Operating limits on the reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using Appendix G of the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, as a guide. These operating limits assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessel shell remote from discontinuities. For the purpose of setting these operating limits the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (65 Edition to Summer 1966 Addenda).

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the RPV. Two types of information are needed in this analysis: 1) A relationship between the change in fracture toughness of the RPV steel and the neutron fluence (integrated neutron flux), and 2) a measure of the neutron fluence at the point of interest in the RPV wall.

A relationship between neutron fluence and change in Charpy V, 30-foot pound transition temperature has been developed for SA302B/SA533 steel based on at least 35 experimental data points as shown in Figure 3.6-2. In turn, this change in transition temperature can be related to a change in the temperature ordinate shown in Figure G 2110-1 in Appendix G of Section III of the Boiler Code.

The neutron fluence at any point in the pressure vessel wall can be computed from core physics data. The neutron fluence can also be measured experimentally on the ID of the vessel wall. At present valid experimental measurements can be made only over time periods of less than five years because of the limitations of the dosimeter materials. This causes no problem because of the exact relationship between thermal power produced and the number of neutrons produced from a given core geometry. A single experimental measurement in a time period of one year can be used to predict the fluence for the life of the plant in terms of thermal power output if no great changes in core geometry are made..

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

The vessel pressurization temperatures at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence and from Figure 3.6-2. For heatup or cooldown and core operation, see curves Nos. 2 and 3 on Figure 3.6-1. During the first fuel cycle, only calculated neutron fluence values can be used. At the first refueling, neutron dosimeter wires which are installed adjacent to the vessel wall can be removed to verify the calculated neutron fluence. As more experience is gained in calculating the fluence the need to verify it experimentally will disappear. Because of the many experimental points used to derive Figure 3.6-2, there is no need to reverify if for technical reasons, but in case verification is required for other reasons, three sets of mechanical test specimens representing the base metal, weld metal and weld heat affected zone metal have been placed in the vessel. These can be removed and tested as required.

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 ≥ 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

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

suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

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

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

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

References

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

3.6.D/4.6.D Relief Valves

To meet the safety basis, 13 relief valves have been installed on the unit with a total capacity of 83.77 percent of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves OPERABLE, results in adequate margin to the code allowable overpressure limit of 1,375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1,375 psig.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief and safety valves are benchtested every second operating cycle to ensure that their setpoints are within the ± 1 percent tolerance. The relief valves are tested in place in accordance with Specification 1.0.MM to establish that they will open and pass steam.

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

References

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)

2. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)

3. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilliland to F. E. Kruesi, August 29, 1973

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, the flow rates in both recirculation loops will be verified by control room monitoring instruments. If the two flow rate values do not differ by more than 10 percent, riser and nozzle assembly integrity has been verified.

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

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

3.6.F/4.6.F Recirculation Pump Operation

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

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

3.6.G/4.6.G Structural Integrity

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

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

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

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in Section 4.6.G.4 to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

References

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

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

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

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

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

ENCLOSURE 4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

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

1.0 INTRODUCTION

By letter dated May 15, 1989, Tennessee Valley Authority (the licensee) requested an amendment to Facility Operating License No. DPR-33, DPR-52 and DPR-58 for the Browns Ferry Nuclear (BFN) Plant, Units 1, 2 and 3. The proposed amendment would change the Technical Specification (TS) 4.6.A.3 and Bases Section 3.6/4.6, Thermal and Pressurization Limitations, to update them in accordance with 10 CFR 50, Appendix H and NRC letter to TVA dated December 2, 1989.

2.0 EVALUATION

The licensee provided a proposed revision to TS 4.6.A.3 to reflect the current revision of ASTM E 185 that their material surveillance program for the reactor vessel must comply in accordance with 10 CFR 50, Appendix H, II.B.1. This section of Appendix H requires that the testing procedures and reporting requirements for each capsule (specimen) withdrawn from the reactor vessel after July 26, 1983 must meet the requirements of ASTM E 185-82 (including a test internal of 6.0 EFPY) to the extent practical for the configuration of the specimens in the capsule. By letter dated December 2, 1988, NRC approved a BFN request to withdraw the first set of surveillance specimens from each reactor vessel at the end of each unit's cycle which most closely approximates 8.0 effective full power years (EFPY) of operation. After the first capsule from each unit has been withdrawn, the subsequent specimens will be tested at 6.0 EFPY interval as required by 10 CFR 50, Appendix H.

This amendment only updates Surveillance Requirement 4.6.A.3 to comply with surveillance program requirement in 10 CFR 50, Appendix H, II.B.1. and is acceptable.

The proposed revision to the Bases Section 3.6/4.6 would provide for the inclusion of the following paragraph to provide clarification as to the agreed upon specimen withdrawal requirements of the material surveillance program for the reactor vessel:

TVA's 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 that 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's letter dated December 2, 1988 stated that BFN could withdraw the first 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.

Revision of the Bases Section, as proposed, provides clarification as to the agreed upon specimen withdrawal program between NRC and TVA as documented in the letter to TVA dated December 2, 1988 and is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

The amendments involve a change to a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 25379) on July 14, 1989 and consulted with the State of Alabama. No public comments were received and the State of Alabama did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: T. Daniels

Dated: August 3, 1989