ıy 18, 1998

Mr. O. J. Zeringue
Chief Nuclear Officer
and Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: AMENDMENT NOS. 251 AND 210TO TO FACILITY OPERATING LICENSE NOS. DPR-52, AND DPR-68: SAFETY/RELIEF VALVE SET POINT TOLERANCE - TECHNICAL SPECIFICATION CHANGE TS-386 (TAC NOS. M97413, AND M97414)

Dear Mr. Zeringue:

The Commission has issued the enclosed Amendment Nos. 251 , and 210 to Facility Operating License Nos. DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant, Units 2 and 3, respectively. These amendments are in response to your application dated December 11, 1996, as supplemented by letter dated November 3, 1997.

The amendments revise the Appendix A Technical Specifications Limiting Safety System Setting (LSSS) 2.2.A relating to the main steam safety/relief valve set points and set point tolerance. Specifically, the revision increases the set point tolerance to ± 3% vice the current ± 11 pound per square inch (approximately 1% of set point value) tolerance. Bases 1.2 and 3.6D/4.6D also are revised.

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

Your December 11, 1996, application requested similar, but not identical, changes for Unit 1. The staff previously informed you of its intention to deny the request for Unit 1. The denial will be documented in separate correspondence.

Sincerely,

/s/

Albert W. De Agazio, Sr. Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-260 and 50-296
Serial No. BFN-98-007

Enclosures: 1. Amendment No. 251 to License No. DPR-52
2. Amendment No. 210 to License No. DPR-68
3. Safety Evaluation

cc w/enclosures: See next page

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

May 18, 1998

Mr. O. J. Zeringue
Chief Nuclear Officer
and Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: AMENDMENT NOS. 251 AND 210TO FACILITY OPERATING LICENSE NOS. DPR-52, AND DPR-68: SAFETY/RELIEF VALVE SET POINT TOLERANCE - TECHNICAL SPECIFICATION CHANGE TS-386 (TAC NOS. M97413, AND M97414)

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The amendments revise the Appendix A Technical Specifications Limiting Safety System Setting (LSSS) 2.2.A relating to the main steam safety/relief valve set points and set point tolerance. Specifically, the revision increases the set point tolerance to ± 3% vice the current ± 11 pound per square inch (approximately 1% of set point value) tolerance. Bases 1.2 and 3.6D/4.6D also are revised.

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

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Sincerely,

Albert W. De Agazio

Albert W. De Agazio, Sr. Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-260 and 50-296
Serial No. BFN-98-007

Enclosures:
1. Amendment No. 251 to License No. DPR-52
2. Amendment No. 210 to License No. DPR-68
3. Safety Evaluation

cc w/enclosures: See next page

Mr. O. J. Zeringue
Tennessee Valley Authority

BROWNS FERRY NUCLEAR PLANT

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 251
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 December 11, 1996, as supplemented by letter dated November 3, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

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

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 251 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

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 18, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 251

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. .An overleaf * page is included to maintain document completeness.

Remove	Insert
1.2/2.2-1	1.2/2.2-1
1.2/2.2-2	1.2/2.2-2
3.6/4.6-30	3.6/4.6-30
3.6/4.6-31	3.6/4.6-31 *

1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY

SAFETY LIMIT

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specifications

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

A. Verify the safety function lift settings of the required S/RVs are within ± three percent of the setpoint as follows:

Number of S/RVs	Setpoint (psig)
4	1105
4	1115
5	1125

Following testing, lift settings shall be within ± one percent.

	Protective Action	Limiting Safety System Setting
B.	Scram--nuclear system high pressure	≤1,055 psig

1.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The safety limits for the reactor coolant system pressure have been selected such that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are not high enough such that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The design pressure (1,250 psig) of the reactor vessel is established such that, when the 10 percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code Section III for pressure transients is added to the design pressure, a transient pressure limit of 1,375 psig is established.

Correspondingly, the design pressures (1,148 for suction and 1,326 for discharge) of the reactor recirculation system piping are such that, when the 20 percent allowance (230 and 265 psi) allowed by USAS Piping Code, Section B31.1 for pressure transients is added to the design pressures, transient pressure limits of 1,378 and 1,591 psig are established. Thus, the pressure safety limit applicable to power operation is established at 1,375 psig (the lowest transient overpressure allowed by the pertinent codes), ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The current cycle's safety analysis concerning the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase is given in the reload licensing report for the current cycle. The reactor vessel pressure code limit of 1,375 psig given in subsection 4.4 of the safety analysis report is well above the peak pressure produced by the overpressure transient described above. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result due to reasonably expected overpressure transients.

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These increased design pressures create a consistent design which assures that, if the pressure within the reactor vessel does not exceed 1,375 psig, the pressures within the piping cannot exceed their respective transient pressure limits due to static and pump heads.

The safety limit of 1,375 psig actually applies to any point in the reactor vessel; however, because of the static water head, the highest pressure point will occur at the bottom of the vessel. Because the

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 two 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 cycle is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their setpoints are within their specified tolerances. 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.

The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION. Overpressure protection is provided during hydrostatic tests by two of the relief valves whose relief setting has been established in conformance with ASME Section XI code requirements. The capacity of one relief valve exceeds the charging capacity of the pressurization source used during hydrostatic testing. Two relief valves are used to provide redundancy.

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. Gilleland to F. E. Kruesi, August 29, 1973

4. 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 $\pm$ 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.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 210
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 December 11, 1996, as supplemented by letter dated November 3, 1997, 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. 210, 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

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 18, 1998

ATTACHMENT TO LICENSE AMENDMENT NO.210

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. An overleaf* page is included to maintain document completeness.

Remove	Insert
1.2/2.2-1	1.2/2.2-1
1.2/2.2-2	1.2/2.2-2
3.6/4.6-30	3.6/4.6-30
3.6/4.6-31	3.6/4.6-31 *

SAFETY LIMIT

1.2 Reactor Coolant System Integrity

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specifications

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 Reactor Coolant System Integrity

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

A. Verify the safety function lift settings of the required S/RVs are within ± three percent of the setpoint as follows:

Number of S/RVs	Setpoint (psig)
4	1105
4	1115
5	1125

Following testing, lift settings shall be within one percent.

Protective Action	Limiting Safety System Setting
B. Scram--nuclear system high pressure	≤1,055 psig

1.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The safety limits for the reactor coolant system pressure have been selected such that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are set high enough such that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The design pressure (1,250 psig) of the reactor vessel is established such that, when the 10 percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code Section III for pressure transients is added to the design pressure, a transient pressure limit of 1,375 psig is established.

Correspondingly, the design pressures (1,148 for suction and 1,326 for discharge) of the reactor recirculation system piping are such that, when the 20 percent allowance (230 and 265 psi) allowed by USAS Piping Code, Section B31.1 for pressure transients is added to the design pressures, transient pressure limits of 1,378 and 1,591 psig are established. Thus, the pressure safety limit applicable to power operation is established at 1,375 psig (the lowest transient overpressure allowed by the pertinent codes), ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

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The safety limit of 1,375 psig actually applies to any point in the reactor vessel; however, because of the static water head, the highest pressure point will occur at the bottom of the vessel. Because the

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 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.

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 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 cycle is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their setpoints are within their specified tolerances. 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.

The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION. Overpressure protection is provided during hydrostatic tests by two of the relief valves whose relief setting has been established in conformance with ASME Section XI code requirements. The capacity of one relief valve exceeds the charging capacity of the pressurization source used during hydrostatic testing. Two relief valves are used to provide redundancy.

References

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

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4. 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-third 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 $\pm$ 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

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NUMBER 251 TO FACILITY OPERATING LICENSE NUMBER DPR-52,

AND AMENDMENT NUMBER 210 TO FACILITY OPERATING LICENSE NUMBER DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

DOCKET NOS 50-260, AND 50-296

1.0 INTRODUCTION

By application dated December 11, 1996,[1] as supplemented by letter dated November 3, 1997,[2] the Tennessee Valley Authority (TVA) proposed an amendment to the Appendix A Technical Specifications (TSs) Limiting Safety System Setting (LSSS) 2.2.A for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3. Specifically, the proposed amendment would allow TVA to increase the allowable main steam safety/relief valve (SRV) set point tolerance to ±3% from the current ± 11 pound per square inch (approximately 1% of set point value) tolerance. Bases 1.2 and 3.6D/4.6D also would be revised. The supplemental submittal did not affect the initial no significant hazards consideration determination.

2.0 BACKGROUND

The Boiling Water Reactor Owners Group (BWROG) has previously submitted the licensing topical report (LTR) NEDC-31753, "BWROG In-Service Pressure Relief Valve Technical Specification Licensing Topical Report,"[3] for staff review. The staff review,[4] dated March 8, 1993, concluded that the LTR provided an acceptable basis for General Electric (GE) BWRs to increase SRV set point tolerances, provided that six plant-specific analysis conditions are satisfied. The staff safety evaluation also concluded that the LTR was acceptable as the basis for the frequency of testing the valves as half the number of valves at least once per 18 months and all within 40 months, with two additional valves tested for each valve found outside the acceptable tolerance.

3.0 DISCUSSION AND EVALUATION

The safety objective of the Nuclear System Pressure Relief System is to prevent over pressurization of the nuclear system. This protects the nuclear system process barrier from failure which could result in the uncontrolled release of fission products. The pressure relief system includes 13 SRVs, arranged into three set point groupings of four valves set at 1105 psig, four valves at 1115 psig and five valves at 1125 psig. The current TSs provide approximately ±1% set point tolerance. The staff safety evaluation of NEDC-31753 approved the increase in SRV set point tolerance to ±3%, provided that six plant-specific conditions are met. These conditions are reviewed below.

Item 1: Transient analyses of all abnormal operational occurrences (AOOs), as described in NEDC-31753P, should be performed utilizing a ±3% set point tolerance for the safety mode of SSVs and SRVs. In addition, the standard reload methodology (or other method approved by the staff) should be used for this analysis.

TVA has stated that the current core Supplemental Reload Licensing Report (SRLR) includes the bounding analyses for AOOs described in NEDC-31753. The analyses were performed utilizing a ±3% set point tolerance. The reload analysis was performed in accordance with the approved GESTAR-II methodology.[5]

Item 2: Analysis of the design basis over pressurization event using the 3% tolerance limit for the SRV set point is required to confirm that the vessel pressure does not exceed the American Society of Mechanical Engineers (ASME) pressure vessel code upset limit.

The current reload licensing report also analyzed the design basis over pressurization event, a main steam isolation valve (MSIV) closure with scram on high reactor power level, utilizing a 3% set point tolerance. The peak vessel pressure for the transient was 1257 psig -- below the ASME limit of 1375 psig.

Item 3: The plant-specific analyses described in Conditions 1 and 2 should assure that the number of SSVs, SRVs, and RVs included in the analyses correspond to the number of valves required to be operable in the technical specification.

Current BFN TSs require that the safety/relief function of 12 of 13 SRVs be operable. This is consistent with the assumptions of the SRLR for the AOOs in Conditions 1 and 2 above.

Item 4: Re-evaluation of the performance of high-pressure systems (pump capacity, discharge pressure, etc.), motor-operated valves, and vessel instrumentation and associated piping must be completed, considering the 3% tolerance limit.

BFN has three systems which are required to inject into the vessel at high pressure conditions: High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC) and Standby Liquid Control (SLC).

The HPCI system is provided to assure that the reactor is adequately cooled to limit fuel cladding temperature in the event of a small break in the nuclear system which does not result in rapid depressurization of the reactor vessel. The HPCI system continues to operate until the vessel pressure is below the pressure at which Low Pressure Coolant Injection or Core Spray can maintain core cooling. The higher system pressure resulting from the increased SRV set point tolerance would result in a small increase in turbine steam flow and steam pressure at both the inlet and outlet of the HPCI turbine, and a corresponding increase in turbine speed. TVA has stated that sufficient margin exists to the steam line high-flow isolation set point and the exhaust line high-pressure trip set point to accommodate the changes in process steam conditions. Also, the HPCI turbine governor is designed to limit turbine speed during operation to less than the overspend trip set point. TVA also has stated that the piping stresses due to pressures that result from an increase in SRV set point tolerance are within the HPCI piping allowable stress limits.

The RCIC system provides make-up water to the reactor vessel during shutdown and vessel isolation conditions to supplement or replace normal make-up sources. The higher system pressure also would result in a small increase in turbine steam flow and steam pressure at both the inlet and outlet of the RCIC turbine, and a corresponding increase in higher turbine speed. TVA has stated that sufficient margin exists to the steam line high flow isolation set point and the exhaust line high pressure trip set point to accommodate the changes in process steam conditions. Also, the turbine governor is designed to limit turbine speed during operation to less than the overspeed trip set point. TVA has stated that the piping stresses due to pressures that result from an increase in SRV set point tolerance are within the RCIC piping allowable stress limits.

The SLC system is a backup system for making the reactor subcritical over the range of operating conditions. The SLC system uses positive displacement pumps which are limited to a discharge pressure of 1425 psig by discharge relief valves. The increased SRV set point tolerance of 3% is, therefore, within the capacity of the SLC system. TVA has also verified that a pressure increase of 3% would not result in over stressing the SLC system piping.

TVA evaluated the performance of motor-operated valves (MOVs) in accordance with Generic Letter (GL) 89-10 for the increased differential pressure loads associated with the proposed SRV set point tolerance. The performance of the MOVs for the proposed 3% tolerance was found to be acceptable, and TVA stated that the master MOV calculations will include the 3% SRV tolerance when they are revised under the Power Uprate Project for increasing the authorized reactor thermal power by 5%.

Item 5. Evaluation of the 3% tolerance on any plant specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.) should be completed.

The current SRLR includes analysis of alternate operating modes, and was performed utilizing a 3% set point tolerance for the SRVs, and was performed in accordance with the staff approved methodology for the alternate operating modes. Currently, BFN is approved for the Extended Load Line Limit, Increased Core Flow, and Final Feedwater Temperature Reduction as alternate operating modes.

Item 6. Evaluation of the effect of the 3% tolerance limit on the containment response during loss of coolant accidents and the hydrodynamic loads on the SRV discharge lines and containment should be completed.

TVA evaluated (Enclosure 5 of Ref. 1) the increased hydrodynamic loading due to SRV actuation with the increased SRV 3% set point tolerance. This evaluation included the effects of the increased SRV set point actuation on containment structural response, steam and water clearing loads on the SRV piping, quenches, supports, submerged structures, and piping attached to the torus. This evaluation determined that the resulting loads are less than 1%

greater than the loads previously used in the Plant Unique Analysis for the controlling load combination where SRV discharge loads are combined with other design loads including dead weight, pressure, thermal, loss-of-coolant accident (LOCA), and earthquake. TVA determined that the resulting combined stresses are within the existing design basis allowable stresses.

TVA's Engineering Report (Enclosure 5 of Ref. 1) enclosed with the December 11, 1996, application states that because SRVs do not open to relieve pressure in the course of a large break LOCA, an increase in the SRV set point tolerance will have no effect on the containment peak accident temperature or pressure. The application further states that for smaller breaks the SRVs may open, but the change has been determined to be negligible, the pressure increase results in a decrease in the specific enthalpy of the steam that is released. Also, the Technical Evaluation Report, prepared by Brookhaven National Laboratory, which was attached with the staff's review[4] of Topical Report NEDC-31753P, indicates that the suppression pool peak temperature would not be affected because the integrated heat load would not change.

Based on the Engineering Report, the proposed amendment would have no significant effect on the peak accident pressure to which the primary containment might be subjected during a design basis accident.

Containment temperature response to a LOCA is of concern with respect to the environmental qualification of electric equipment in containment. The BFN primary containment design temperature of 281°F was based on a double-ended guillotine break of the Recirculation System piping (DBA-LOCA). It was later discovered that a small-break LOCA may produce a greater containment temperature which can be limited by manual operator initiation of the containment spray system within 30 minutes. Based on the availability of containment spray to limit the primary containment post-accident temperature, the proposed amendment would not introduce any new containment temperature concerns.

The proposed amendment will allow TVA to increase the allowable SRV set point tolerance from approximately ±1% to ±3%. The BWROG has previously submitted NEDC-31753, "BWROG In-service Pressure Relief Valve Technical Specification Licensing Topical Report," for staff review. The staff review concluded that the LTR provided an acceptable basis for GE BWRs to relax SRV set point tolerances, provided certain plant-specific analyses are provided. TVA has provided these analyses for BFN units 2 and 3, and the results of these analyses are acceptable to the staff. Therefore, the changes are acceptable based on the conditions as given in the staff Safety Evaluation[4] for NEDC-31753.

Furthermore, the staff has determined the proposed increase in SRV set point tolerance from 1% to 3% will not result in an unacceptable increase in containment DBA-LOCA pressure/temperature loads. This determination is based on the information provided in the TVA Engineering Report which addresses the containment response analysis requirement identified in the staff's March 8, 1993, Safety Evaluation for NEDC-31753. Therefore, the proposed change to Limiting Safety System Setting 2.2.A is acceptable. The change requires verification that the lift settings of the safety/relief valves are within ±3% of the specified set points.

TVA has proposed to revise Bases 1.2 to reference the *reload licensing report* vice the *reload licensing submittal* and has updated a referenced section in the safety analysis report. Additionally, TVA has proposed changes to Bases 3.6D/4.6D to refer to testing on a cycle basis and to delete reference to the specific set point tolerance. The staff has no objection to these proposed changes.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 a surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 2194). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: May 18, 1998

7.0 REFERENCES

1. Letter from T. E. Abney (TVA) to NRC, "Browns Ferry Nuclear Plant - Units 1, 2, and 3 - Technical Specification (TS) 386 - Proposed change to Safety/Relief Valve (S/RV) set point Requirements for Reactor Coolant System Integrity, TS 2.2.A," December 11, 1996.

2. Letter from T. E. Abney (TVA) to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Request for Additional Information Regarding Increase in Main Steam Safety/Relief Valve (S/RV) Set point Tolerance," November 3, 1997.

3. NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," February 1990.

4. Letter from A. C. Thadani (NRC) to C. L. Tully (BWROG), "Acceptance for Referencing of Licensing Topical Report NEDC-31753P, 'BWROG In-Service Pressure Relief Technical Specification Licensing Topical Report,'" March 8, 1993.

5. NEDE-24011-P-A-11, "General Electric Standard Application for Reactor Fuel, GESTAR II," and NEDE-24011-P-A-11-US, "GESTAR II U. S. Supplement,"