TVA

Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609

December 11, 1996

TVA-BFN-TS-386

10 CFR 50.4
10 CFR 50.90
10 CFR 50.91

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - 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

In accordance with the provisions of 10 CFR 50.4 and 50.90, TVA is submitting a request for an amendment (TS-386) to licenses DPR-33, DPR-52, and DPR-68 to increase the allowed main steam S/RV set point tolerance specified in Limiting Safety System Setting Specification 2.2.A. The current 2.2.A specifies a ± 11 pounds per square inch tolerance (approximately 1% of set point value) for each S/RV group. This proposed change would provide for a set point tolerance of ± 3%.

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The Boiling Water Reactor Owners Group (BWROG) has previously submitted Licensing Topical Report NEDC-31753P, "BWROG In-Service Pressure Relief Valve Technical Specification Licensing Topical Report," to NRC for review. TVA was a participant in this BWROG activity. The topical report documents the generic evaluation to support the modification of the set point tolerance of main steam S/RVs.

NRC subsequently issued a Safety Evaluation Report (SER) (Reference: Letter from A. C. Thadani, NRC, to C. L. Tully, BWROG, dated March 8, 1993) which concluded that NEDC-31753P provided an acceptable basis for General Electric Boiling Water Reactors (BWR) to relax the in-service pressure set point tolerances for S/RVs. As stated in the SER, licensees choosing to implement TS changes were also required to provide certain plant specific analyses. These plant specific analyses are included in the enclosures for Unit 2.

The corresponding Unit 3 reactor transient analysis is being performed for the next core reload cycle. Required Unit 1 analyses will be performed prior to return of that unit to service.

This proposed change is consistent with proposed TS 362, dated September 6, 1996, which submitted TVA's proposed conversion of the custom BFN TS to be consistent with NUREG-1433, Revision 1, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/4)." Also, similar TS changes involving S/RV set point tolerances have been recently approved for other BWRs (e.g., Grand Gulf, LaSalle, and Millstone).

Enclosure 1 to this letter provides the description and evaluation of the proposed change including a summary of the BFN plant specific analyses. This enclosure also includes TVA's determination that the change does not involve a significant hazards consideration, and is excluded from environmental review.

Enclosure 2 contains marked-up copies of the appropriate Unit 1, 2, and 3 TS pages to show the proposed change. Enclosure 3 forwards the corresponding revised TS pages which incorporate the proposed change. The current (reload 8, cycle 9) Unit 2 reload licensing report is contained in Appendix N of the Updated Final Safety Analysis Report and a copy has been included in Enclosure 4. Enclosure 5 is an Engineering Report which provides additional detail on plant specific considerations and evaluations.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change qualifies for a categorical exclusion pursuant to the provisions of 10 CFR 51.22(c)(9) for environmental review. The BFN Plant Operations Review Committee and the BFN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of BFN Units 1, 2, and 3

in accordance with the proposed change will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Alabama State Department of Public Health.

TVA requests that the revised Units 1 and 2 TS be made effective within 30 days of NRC approval. The Unit 3 TS should be made effective prior to the restart from the pending refueling outage in March 1997 or within 30 days of NRC approval, whichever is later.

If you have any questions about this change, please contact me at (205) 729-2636.

Sincerely,

T. E. Abney

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosures
cc: see page 4

Subscribed and sworn to before me
on this 11th day of Dec. 1996.
Barbara A. Blanton
Notary Public

My Commission Expires My Commission Expires 10/06/98

Enclosures
cc (Enclosures):

Mr. Johnny Black, Chairman
Limestone County Commission
310 West Washington Street
Athens, Alabama 35611

Mr. Mark S. Lesser, Branch Chief
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

NRC Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611

Mr. Joseph F. Williams, Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

Dr. Donald E. Williamson
State Health Officer
Alabama State Department of Public Health
434 Monroe Street
Montgomery, Alabama 36130-3017

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, and 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-386
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I. DESCRIPTION OF THE PROPOSED CHANGE

TVA is requesting a change to Limiting Safety System Setting (LSSS) 2.2 to revise the "as-found" tolerance for the main steam safety/relief valves (S/RV) set points. The specific change is described below:

1. Current TS LSSS 2.2.A (Page 1.2/2.2-1). All units TS are the same except that Unit 3 TS does not have the column headers:

Protective Action	Limiting Safety System Setting
A. Nuclear system relief valves open--nuclear system pressure	1,105 psig ± 11 psi (4 valves)
	1,115 psig ± 11 psi (4 valves)
	1,125 psig ± 11 psi (5 valves)

Proposed TS LSSS 2.2.A:

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.

2. Current Units 1, 2, and 3 Bases 3.6.D/4.6.D, Relief Valves (Page 3.6/4.6-30).

> ... 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...

Proposed Bases 3.6.D/4.6.D, Relief Valves:

> ... 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...

3. Current Unit 1 Bases 1.2, Reactor Coolant System Integrity (Page 1.2/2.2-2)

> ...The current cycle's safety analysis concerning the most severe operational transient resulting directly in a reactor coolant system pressure increase is given in reference 5.
>
> The reactor vessel pressure code limit of 1,375 psig given in subsection 4.2 of the safety analysis report...

Proposed Unit 1 Bases 1.2, Reactor Coolant System Integrity (Page 1.2/2.2-2).

> ...The current cycle's safety analysis concerning the most severe 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...

Current Unit 1 Bases 1.2, Reactor Coolant System Integrity (Page 1.2/2.2-3)

> ...1. Plant Safety Analysis (BFNP FSAR Section 14.0)

Proposed Unit 1 Bases 1.2, Reactor Coolant System Integrity (Page 1.2/2.2-3)

> ...1. Plant Safety Analysis (BFNP FSAR Sections 14.4 and Appendix N)

Current Unit 1 Bases 1.2, Reactor Coolant System Integrity (Page 1.2/2.2-4)

> ...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...

Proposed Unit 1 Bases 1.2, Reactor Coolant System Integrity (Page 1.2/2.2-4)

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Current Unit 1 Bases 3.6.D/4.6.D, Relief Valves (Page 3.6/4.6-30).

> ...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...

Proposed Unit 1 Bases 3.6.D/4.6.D, Relief Valves (Page 3.6/4.6-30).

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4. Current Units 2 and 3 Bases 1.2, Reactor Coolant System Integrity (Page 1.2/2.2-2).

> ...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 submittal for the current cycle. The reactor vessel pressure code limit of 1,375 psig

given in subsection 4.2 of the safety analysis report...

Proposed Units 2 and 3 Bases 1.2, Reactor Coolant System Integrity (Page 1.2/2.2-2).

...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...

The above changes will make the affected bases sections the same for all three units.

II. REASON FOR THE PROPOSED CHANGE

The Boiling Water Reactor Owners Group (BWROG) previously submitted Licensing Topical Report NEDC-31753P, "BWROG In-Service Pressure Relief Valve Technical Specification Licensing Topical Report", to NRC for review. The topical report documents the generic evaluation to support the modification of the main steam S/RV set point tolerance from ± 1% to ± 3%. BFN was a participant in this BWROG activity. NRC subsequently issued a Safety Evaluation Report (SER) (Reference: Letter from A. C. Thadani, NRC, to C. L. Tully, BWROG, dated March 8, 1993) which concluded that NEDC-31753P provided an acceptable basis for General Electric Boiling Water Reactors (BWR) to relax the in-service pressure set point tolerances for S/RVs to ± 3%. As stated in the SER, licensees choosing to implement the subject TS changes were also required to provide certain plant specific analyses.

The BFN plant specific analyses specified in the NRC SER have been completed for Unit 2. The required Unit 3 analyses are complete except for the core reload licensing report which is currently being performed for the next operating cycle (March 1997). Corresponding Unit 1 analyses will be performed prior to return of that unit to service.

This change will result in reduced testing for S/RVs which fail to operate within ± 1%, but are within the ± 3% tolerance while the plant is operating. For this situation, the revised TS would allow the affected S/RV to be considered operable. The proposed TS change explicitly requires that any valve found performing outside the ± 1% tolerance during bench-testing be returned to within a ± 1% tolerance prior to reuse.

BFN exclusively uses Target Rock Two-Stage valves for main steam S/RVs. Target Rock Two-Stage S/RVs have exhibited a generic set point drift tendency which has made the 1% set point tolerance difficult to consistently achieve during testing following operating service in the field. The BWROG continues to have an active effort in this area and is sponsoring design and materials initiatives to improve the overall set point performance of Target Rock S/RVs. Use of an increased tolerance based on the BWROG industry precedent is one element of the remedy and will improve the overall test results by providing added margin in the test criteria.

This proposed change is consistent with TS 362, submitted September 6, 1996, which is TVA's proposed conversion of the custom BFN Technical Specifications to Improved Standard Technical Specifications format per NUREG-1433, Revision 1, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/4)." The technical justification presented in this submittal is also applicable to the proposed ISTS submittal with regard to S/RV set point tolerances.

III. SAFETY ANALYSIS

SYSTEM DESCRIPTION

As discussed in the BFN Updated Final Safety Analysis Report (UFSAR), Section 4.4, Nuclear System Pressure Relief System, the safety objective of the Nuclear System Pressure Relief System is to prevent overpressurization of the nuclear system. This protects the nuclear system process barrier from failure which could result in the uncontrolled release of fission products.

The Nuclear System Pressure Relief System includes 13 combination safety-relief valves which are located on the main steam lines within the drywell between the reactor vessel and the steam line flow restrictors. The S/RVs are distributed among the four main steam lines so that a single accident cannot completely disable a safety, relief, or automatic depressurization function. The S/RVs, which discharge to the suppression pool, provide three main protection functions:

1. Overpressure relief operation. The valves are opened in a self-actuated relief mode by process steam pressure to limit the pressure rise during certain operational reactor transients.

2. Overpressure safety operation. The valves open in a self-actuated relief mode by process steam pressure to prevent nuclear system vessel and appurtenant piping overpressurization.

3. Depressurization operation. Six of the 13 S/RVs function as Automatic Depressurization System (ADS) valves and are opened by automatically operated devices as part of the Emergency Core Cooling System (ECCS), when and if required, for small breaks in the nuclear system process barrier. All 13 valves can also be remotely operated in a manual mode by operator action. The ability to open the S/RVs automatically in an ADS mode or manually by remotely operated devices is not affected by this proposed revision to the S/RVs set point tolerances.

The S/RVs are designed, constructed, and marked with data in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9, 1965 edition, and in accordance with ASME Boiler and Pressure Vessel Code, Section III, 1968 edition and addenda through summer 1970. Set point tolerance (pressure at which valve "pops" wide open) is in accordance with ASME Boiler and Pressure Vessel Code, Section I, paragraph PG-72(c).

The 13 S/RVs are arranged into three set point groupings as specified in TS 2.2.A. The groupings are four valves at 1,105 pounds per square inch gauge (psig), four valves at 1,115 psig, and five valves at 1,125 psig. Existing TS provide a ± 11 psi (approximately 1% of range) set point tolerance. BFN exclusively uses Target Rock Two-Stage valves.

SAFETY EVALUATION

Licensing Topical Report NEDC-31753P was submitted to NRC by the BWROG to request generic approval of an increased tolerance for the S/RV safety function lift set points. NRC subsequently issued a SER on the NEDC which indicated that a generic change of safety-relief valve set point "as-found" tolerances to ± 3% was acceptable, provided that certain plant specific analyses were performed. The BFN plant specific analyses specified in the NRC SER have been completed for Units 2 and 3, except for the Unit 3 reactor transient analysis which is being performed for the next core reload cycle. Similar Unit 1 analyses will be performed prior to return of that unit to service.

ANSI/ASME OM-1-1981, "Requirements for In-Service Performance Testing of Nuclear Plant Pressure Relief Devices," states in paragraph 1.3.3.1.5(b) that "Any valve exceeding its stamped set pressure by 3% or greater shall be repaired or replaced..." Subsection IWV-3510 (Safety Valve and Relief Valve Tests) of Section XI states that corrective action shall be in accordance with the requirements of ANSI/ASME OM-1-1981 for valves not meeting the acceptance criteria of OM-1-1981 (+3%). Thus, the industry position through use of OM-1-1981 supports the proposed S/RV set point tolerance change to ± 3%.

The proposed TS change does not involve physical changes to the S/RVs or involve different modes of operation. Also, the TS amendment does not alter the frequency of verifying the S/RVs lift set points or the number of S/RVs required to be operable. The S/RVs lift set points will still be set within a tolerance of ± 1%, but the set points will be tested to within ± 3% to determine acceptance or failure of the "as-found" valve lift set point. Valves with set points found outside of the ± 1% tolerance during bench-testing will be reset to within the ± 1% or replaced by valves with set points set to that tolerance.

A summary of the results of the plant specific evaluations and safety conclusions is provided below. Added detail is contained in the Engineering Report in Enclosure 5.

SUMMARY OF SAFETY EVALUATION REPORT (SER) PLANT SPECIFIC ANALYSES (PSA) RESULTS

NRC SER PSA ITEM 1

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

RESPONSE TO NRC SER PSA ITEM 1

The current Unit 2 core reload licensing analysis report (reload 8, cycle 9) includes the bounding analyses for the anticipated operational occurrences described in NEDC-31753P and were performed utilizing a ± 3% set point tolerance. This reload analysis was performed in accordance with NRC approved methodology (General Electric Licensing Topical Report NEDE-24011-P-A) as referenced in TS 6.9.1.7.b for the Core Operating Limits Report.

A current copy of the Unit 2 reload licensing analysis is provided in enclosure 4. For this Unit 2 reload cycle, as expected, the generator load reject without bypass valve operation event is the limiting operational transient and the analysis shows acceptable core thermal limits (Critical Power Ratio) results.

The corresponding Unit 3 core reload analyses for the next operating cycle (starts in March 1997) are in progress and likewise will also use the ± 3% set point tolerance. Prior to the return of Unit 1 to service, the same analyses will be performed.

NRC SER PSA ITEM 2

Analysis of the design basis overpressurization event using

the 3% tolerance limit for the safety/relief valve set point is required to confirm that the vessel pressure does not exceed the ASME pressure vessel upset limit.

RESPONSE TO NRC SER PSA ITEM 2

The current Unit 2 reload licensing report also analyzed the design basis pressurization event (Main Steam Isolation Valve (MSIV) Closure with flux scram) utilizing a + 3% set point tolerance and shows that the peak vessel pressure is well within ASME pressure vessel limits. Section 12 of the reload analysis provides the numerical results and the transient response is shown in a graphical format in Figure 15. The peak steam line pressure was 1224 psig and the peak vessel pressure was 1257 psig.

The corresponding Unit 3 core reload licensing analysis for the next operating cycle (March 1997) is in progress and will also use a + 3% set point tolerance for the design basis overpressurization event. Prior to the return of Unit 1 to service, the same analysis will be performed.

Previous TVA calculations also indicate there is additional margin available in terms of defining an upper limit for an S/RV set point tolerance. Specifically, TVA transient calculations for the Unit 2 Cycle 6 core using the RETRAN-02 analysis program indicated that a set point tolerance of + 10% with 4 out of 13 S/RVs assumed inoperable would still meet design basis for the overpressure requirements. Since the pressurization transients are not very sensitive to changes in core reloads, it can be concluded that further relaxation in S/RV set point tolerances is feasible without encroaching on core thermal or pressure limits.

<u>NRC SER PSA ITEM 3</u>

The plant specific analyses described in Items 1 and 2 should assure that the number of spring safety valves, safety/relief valves, and relief valves included in the analyses correspond to the number of valves required to be operable in the Technical Specifications.

RESPONSE TO NRC SER PSA ITEM 3

The number of S/RVs assumed operable in the reload analysis for the events referenced in Items 1 and 2 above is the same as that specified in TS 3.6.D, Relief Valves. Namely, there are 13 total S/RVs, and 12 S/RVs are assumed to be operable in the current Unit 2 reload licensing report and in the Unit 1 and 3 reload analyses. This is a standard input assumption for the core reload analyses and a change to TS 3.6.D would be necessary to take credit for a different number of inoperable S/RVs.

NRC SER PSA 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.

RESPONSE TO NRC SER PSA ITEM 4

BFN has three systems which are required to inject to the vessel at high pressure conditions. A discussion of the capability of each system to operate at a slightly higher pressure (1105 psig +3% as opposed to 1105 psig +1%) is provided below. These three high pressure systems are:

1. High Pressure Coolant Injection (HPCI),
2. Reactor Core Isolation Cooling (RCIC); and,
3. Standby Liquid Control (SLC).

Each of these systems was evaluated for the effects of operating at a higher vessel pressure as follows. Added detail is provided in the Engineering Report (Enclosure 5).

1. HIGH PRESSURE COOLANT INJECTION

   HPCI 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 and loss of coolant which does not result in rapid depressurization of the reactor vessel. HPCI also supplies high pressure makeup water during events that involve loss of feed water or MSIV closure.

   The HPCI system permits the nuclear plant to be shut down, while maintaining sufficient reactor vessel water inventory until the reactor vessel is depressurized. The HPCI continues to operate until the reactor vessel pressure is below the pressure at which Low Pressure Coolant Injection operation or Core Spray System operation maintains core cooling.

   HPCI is designed to deliver a flow rate of 5,000 gallons per minute (gpm) into the reactor vessel with the reactor vessel pressure at 1120 psig. HPCI is a turbine driven pump with variable speed control which will automatically regulate flow to the reactor at 5,000 gpm. HPCI has excess capacity and is capable of delivering rated flow at the slightly elevated reactor pressure.

   A higher reactor pressure would result in a small increase in turbine steam flow and steam pressure at both the inlet and outlet of the HPCI turbine, and in a slightly higher turbine speed. Sufficient margin exists to the steam line high flow isolation set point

and the exhaust line high pressure trip set point to accommodate these small changes in process steam conditions. Also, the turbine governor limits turbine speed during operation to less than the overspeed trip set point.

HPCI piping allowable pressures are well above the pressures that result from an increase in S/RV set point tolerance.

Therefore, it is concluded HPCI operation would not be adversely affected by an elevated pressure.

2. REACTOR CORE ISOLATION COOLING

RCIC provides make-up water to the reactor vessel during shutdowns and isolations from the main heat sink to supplement or replace the normal make-up sources, and operates automatically in time to obviate any need for use of the ECCS.

RCIC is designed to deliver a flow rate of 600 gpm into the reactor vessel with vessel pressure at 1120 psig. RCIC is a turbine driven pump with variable speed control which will automatically regulate flow to the reactor at 600 gpm. RCIC has excess capacity and is capable of delivering rated flow at the slightly elevated reactor pressure.

A higher reactor pressure would result in a small increase in turbine steam flow and pressure at both the inlet and outlet of the turbine, and in a slightly higher turbine speed. Sufficient margin exists to the steam line high flow isolation set point and the exhaust line high pressure trip set point to accommodate these small changes process steam conditions. The turbine governor limits turbine speed during operation to less than the overspeed trip set point.

RCIC piping allowable pressures are well above the pressures that result from an increase in S/RV set point tolerance.

Therefore, it is concluded RCIC operation would not be adversely affected by an elevated pressure.

3. STANDBY LIQUID CONTROL

The safety objective of the SLC system is to provide a backup method, which is independent of the control rods, to make the reactor subcritical over its full range of operating conditions.

The SLC system uses positive displacement pumps which

are limited in discharge pressure by discharge relief valves at 1425 psig. The increased set point tolerance of 3% (reactor vessel pressure of 1138 psig) is well within the flow capacity of the system and pressure rating of the piping.

MOTOR OPERATED VALVES (MOV)

As described in the Technical Evaluation Report (TER) prepared as part of the NRC'S evaluation for NEDC-31753P, consideration should be given to testing MOVs exposed to reactor pressure at higher differential pressures. Because MOV dynamic testing is done at the highest differential pressure achievable under normal operational configurations, dynamic testing parameters are unaffected by the S/RV setting tolerance increases.

MOV operator settings for static testing are also based on the calculated maximum expected differential pressure values (as one of the input parameters for determining required settings). Therefore, MOVs capability calculations for MOVs which are potentially affected by the change in S/RV set point tolerance were evaluated for the effect of an elevated pressure of 1138 psig (1105 psig + 3%). These MOVs were determined to have sufficient margin to perform their function.

VESSEL INSTRUMENTATION

As described in NEDC-31753P and the TER, consideration should be given to the effects on vessel instrumentation induced by the safety/relief valve safety setting tolerance increase. Instruments potentially effected by the proposed change were evaluated for the effects on pressure boundary integrity, instrument calibration, instrument scaling calculations and instrument set point/uncertainty calculations. The evaluation determined that there is no adverse impact on plant instrumentation as a result of the proposed set point tolerance change.

<u>NRC SER PSA ITEM 5</u>

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

RESPONSE TO NRC SER PSA ITEM 5

The current Unit 2 reload licensing report includes a analysis for the alternate operating modes, and was performed utilizing a ± 3% set point tolerance for the S/RVs. The core reload analysis was performed in accordance with NRC approved methodology for the alternate operating modes. The operating cycle flexibility options currently used at BFN Unit 2 are shown on pages 7 and 8 of

the reload analysis. These include Extended Load Line Limit, Increased Core Flow, and Final Feedwater Temperature Reduction.

The corresponding Unit 3 reload licensing analyses for the next operating cycle (March 1997) is in progress and, likewise, will also use the ± 3% set point tolerance for the flexibility options selected. Prior to the return of Unit 1 to service, the same corresponding analyses will be performed.

NRC SER PSA 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 safety/relief valve discharge lines and containment should be completed.

RESPONSE TO NRC SER PSA ITEM 6

CONTAINMENT RESPONSE DURING LOSS OF COOLANT ACCIDENTS (LOCA)

The most limiting event in terms of peak containment pressure and temperature, and peak suppression pool temperature is the design basis LOCA accident which is a double ended guillotine break of a recirculation line. Relaxation of the S/RV set point tolerance has no effect on this event because the vessel depressurizes without any S/RV valve actuations. Therefore, there is no impact on the LOCA peak containment pressure and temperature, and on the peak LOCA suppression pool temperature. The set point tolerance change has been determined to have negligible impact on containment responses for small break and intermediate break LOCAs.

S/RV DISCHARGE PIPING LOADS AND CONTAINMENT HYDRODYNAMIC LOADS

The following effects of increased S/RV set point tolerance have been evaluated for Units 2 and 3.

a. Containment structural response from S/RV hydrodynamic loads.

b. Steam and water clearing loads on the S/RV lines, quenchers, and supports.

c. Increases in hydrodynamic loads on submerged structures.

d. Effects on piping attached to the torus.

Calculations demonstrate that S/RV loads resulting from the proposed + 3% set point increase are less than 1% higher

than the loads used in design. When S/RV loads are combined with other design basis loads including dead weight, pressure, thermal, LOCA, and earthquake, the total load increase is negligible and affected components meet design basis requirements.

NRC SAFETY EVALUATION REPORT (SER) CONCLUSIONS AND LIMITATIONS (C&L)

The SER for NEDC-31753P provided four conclusions and limitations. The TVA response to each is as follows:

NRC SER C&L ITEM A

A generic change of valve set point tolerance to ± 3% is acceptable.

RESPONSE TO NRC SER C&L ITEM A

TVA's proposed TS change is in agreement with this tolerance.

NRC SER C&L ITEM B

The staff concludes that the philosophy of an upper limit is not acceptable as a means to further reduce the number of Licensee Event Reports (LERs), and that an evaluation to determine the necessity for filing an LER must be made for set points when drift outside ± 3% set point tolerance is found.

RESPONSE TO NRC SER C&L ITEM B

TVA will continue to evaluate nuclear system S/RV test results above their limiting safety system setting (including tolerances) for reportability in accordance with 10 CFR 50.72 and 10 CFR 50.73 whenever drifts are observed outside ± 3% during testing.

NRC SER C&L ITEM C

The recommendation to modify the Technical Specifications to classify a valve as operable with the set point of a valve(s) outside of the three percent tolerance is not technically justified by the Licensing Topical Report or by the supplemental data, and therefore, not acceptable.

RESPONSE TO NRC SER C&L ITEM C

TVA agrees with this position. The proposed TS uses a ± 3% tolerance.

NRC SER C&L ITEM D

Modification of the current requirement for spring safety

valves (SSV) and, safety/relief valve testing to require testing of one-half of the valves at least once per 18 months and all within 40 months is acceptable. Plant specific Technical Specification changes to raise the allowable drift tolerance to ± 3% must include the requirement that additional testing be conducted if failures are experienced; i.e., two additional valves for each valve found with a set point outside the three percent tolerance. In all cases the valve set point will be restored to within ± 1% prior to plant startup. Such testing requirements are consistent with the existing testing requirements.

RESPONSE TO NRC SER C&L ITEM D

Current TS 4.6.D requires one-half of the S/RVs be removed and bench-checked or replaced with a bench-checked valve each operating cycle, and that all 13 valves be checked or replaced by every second cycle. No TS changes are being proposed to this requirement.

In addition, the BFN administrative procedure for implementing the ASME Section XI In-Service Testing of Pumps and Valves program requires the S/RVs be periodically bench-checked in accordance with IWV-3510 and NUREG-1482. IWV-3510 requires the testing and frequency be in accordance with ANSI/ASME OM-1-1981. OM-1 requires that two additional valves be tested for each valve that fails to meet the valve lift pressure by + 3% or greater. If any of the additional valves fail to meet + 3%, OM-1 requires all of the same type and manufacture to be tested. The proposed TS change explicitly requires that the S/RV set point be reestablished at ± 1% for all S/RVs prior to reinstallation.

Therefore, the requested TS change and site test procedures are consistent with the SER position.

IV. <u>NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION</u>

TVA has concluded that operation of BFN Units 1, 2, and 3 in accordance with the proposed change to the TS does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

A. <u>The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

TVA is proposing a change to the "as-found" tolerances for the S/RV set points. This proposed TS amendment does not alter the frequency of verifying the S/RV

lift set points, or the number of S/RVs required to be operable. The amendment does not involve physical changes or modifications to the S/RVs, or change the operating mode or safety function of the S/RVs. The safety lift set points will still be required to be set within a tolerance of ± 1% following testing.

S/RV actuation is not a precursor to any design basis accident analyzed in the BFN UFSAR. Therefore, this change does not increase the probability of any previously evaluated accident.

Generic considerations related to the set point tolerances were addressed in NEDC-31753P and previously reviewed by NRC. In accordance with the NRC SER on utilizing the NEDC results, certain plant specific evaluations were performed to support the proposed change. Specifically, the current Unit 2 reload licensing report includes the transient analyses for the anticipated operational occurrences and the limiting overpressurization transient utilizing the ± 3% S/RV set point tolerance and were performed in accordance with NRC approved methods. The alternate operating modes were also included in the reload licensing report. These analyses concluded there is adequate margin to design core thermal limits and pressure limits for the reactor vessel. The corresponding Unit 3 core reload licensing report for the next operating cycle (starts in March 1997) is in progress and will also use the ± 3% S/RV set point tolerance. Prior to the return of Unit 1 to service, the same reload analysis will be performed. Similar results to those for Unit 2 are expected.

The operation of high pressure injection systems have been determined not to be adversely affected by the proposed change. LOCA response, containment hydrodynamic loads, pump and valve performance, and instrumentation performance were likewise satisfactorily evaluated. Therefore, this proposed change does not significantly increase the consequences of any previously evaluated accident.

B. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed change does not involve a modification to plant equipment. No new failure modes are introduced. Plant systems will continue to function and no new system interactions are introduced by this proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. <u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>

The proposed change has been analyzed in accordance with NRC approved methodology and the margins of safety for the design basis accidents and transients analyzed in Chapter 14 of the BFN UFSAR have not been significantly reduced. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

V. <u>ENVIRONMENTAL IMPACT CONSIDERATION</u>

The proposed change does not involve a significant hazards consideration, a change in the types of, or increase in, the amounts of any effluents that may be released off-site, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.