R92 961113 878

QA Record

# BROWNS FERRY NUCLEAR PLANT

# ENGINEERING REPORT

## SAFETY/RELIEF VALVES
## SAFETY FUNCTION LIFT SETPOINT
## TOLERANCE RELAXATION
## SUMMARY REPORT

| | | | |
|---|---|---|---|
| Prepared By: | John D. McCoy | Date | 8 NOV 96 |
| Reviewed By: | [illegible] | Date | 11-8-96 |
| Reviewed By: | JR Class | Date | 11-12-96 |
| Reviewed By: | [illegible] | Date | 11-12-96 |
| Reviewed By: | John Lee [illegible] | Date | 11-12-96 |
| Reviewed By: | D. L. [illegible] | Date | 11/12/96 |
| Approved By: | H L Williams / [illegible] | Date | 11/12/96 |

CONTENTS

## EXECUTIVE SUMMARY

This report presents the results of the Browns Ferry Nuclear Plant (BFN) plant-specific evaluations (References 4.1, 4.2, and 4.3) to support increasing the safety/relief valves (SRV) setpoint tolerance from ± 1% to ± 3%. The current requirements for the Automatic Depressurization System operation are maintained. Additionally, the existing SRV safety mode setpoints are retained, along with the requirement to reset the valves to within ± 1% before returning the valves to service after testing. The performance change is intended to minimize the impact to plant operations from potential pressure relief system related problems due to SRV operability considerations while maintaining acceptable levels of safety. The change is expected to provide benefit in the areas of Licensee Event Report (LER) reduction, and a reduction of required SRV testing and maintenance, along with associated costs. Based on the plant specific evaluation of References 4.1, 4.2 and 4.3, and the generic evaluation contained in Reference 4.4, Licensing Topical Report (LTR) NEDC-31753P, it is concluded that the proposed change in setpoint tolerance results in plant and system performance that is within BFNs design and licensing basis.

# 1.0 INTRODUCTION

## 1.1 BACKGROUND

The BWR Owners Group (BWROG) submitted Licensing Topical Report (LTR) NEDC-31753P (Reference 4.4) to the NRC which provided technical justification for the SRV safety setpoint tolerance requirements. The LTR presented the justification for the change in two parts, a generic evaluation for applicable areas, and a plant-specific evaluation. NRC approved the request for a change in SRV setpoint tolerance to ± 3% per Reference 4.5, with the requirements that certain plant-specific issues be addressed by licensees adopting the change. BFN is proposing to implement a Technical Specification change for the SRV setpoint tolerance relaxation, consistent with the requirements presented in Reference 4.5, and has completed the plant-specific evaluations (References 4.1, 4.2, and 4.3) required to support the change.

The vessel overpressure protection system at BFN is designed to satisfy the requirements of Section III, Nuclear Power Plant Components, of the ASME Boiler and Pressure Vessel Code. The safety/relief valves (total of 13) which are a part of the vessel overpressure protection system are sized to limit the primary system pressure, including transients, to the requirements of the ASME Code, Section III. The nuclear pressure relief system has been designed (Reference 4.6):

- To prevent overpressurization of the nuclear system that could lead to the failure of the reactor coolant pressure boundary, and

- To provide automatic depressurization for small breaks in the nuclear system occurring with maloperation of the high pressure core injection (HPCI) system so that the low pressure coolant injection (LPCI) and the core spray (CS) systems can operate to protect the fuel barrier.

BFN utilizes two stage Target Rock pilot operated safety/relief valves (Model 7567F) which open in the safety/relief mode in response to system pressure actuating the pilot assembly, which then opens the reverse-seated, hydraulically-actuated angle globe valve. A remotely-controlled air operator is fitted to the pilot stage assembly to provide selective operation of the valve at system pressure other than set pressure. This is a diaphragm-type, pneumatic actuator which must be actuated to open the valve.

## 1.2 PRESENT REQUIREMENTS

Technical Specifications (Reference 4.7) require that the safety/relief function of 12 SRVs be operable. The nominal opening setpoints and tolerances for the valves are as follows:

| Number of SRVs | Safety/Relief Setting, psig |
|---|---|
| 5 | 1125 ± 11 |
| 4 | 1115 ± 11 |
| 4 | 1105 ± 11 |

The current MSRV setpoint tolerance listed in the BFN Technical Specifications, is based on original GE specifications (the GE specifications were based, in general, on the existing industry practices and ASME Code, Section III construction requirements at that time. Note: BFN was not constructed to ASME Section III.). Operability determinations of the valves is based in part on Technical Specification surveillance test acceptance criteria for an as-found setpoint tolerance of 1%. Exceeding the Technical specification tolerance on an as-found test requires performance of evaluations to determine if the plant has operated in an unanalyzed condition in addition to reportability requirements. In addition, test failures during scheduled Code testing based on the acceptance criteria would result in testing additional valves in accordance with Code requirements. Additional/expanded testing is tied to failures during scheduled Code testing; OM-1 does not require expanded testing associated with operational failures. Note also, that the Code requires SRV testing to be completed prior to startup if less than the full complement of valves were replaced. (As a matter of choice, BFN routinely replaces all 13 SRVs each refueling outage to preclude potential outage schedule impact due to additional valve testing requirements as a result of failure to meet the setpoint requirements. An additional benefit is realized by replacing all 13 SRVs in that BFN then has 12 months to perform the as-found testing.)

The BFN In-Service Testing (IST) Program is currently performed in accordance with the requirements of the ASME Code, Section XI, 1986 Edition. The SRVs at BFN are classified as Category C valves according to the BFN IST Program. The SRV test requirements are addressed in the BFN IST Program. Required SRV test methods and frequencies will not be affected by the proposed change in SRV setpoint tolerance. The Category C test requirements are satisfied by performing bench testing (including set pressure, seat leakage and stroke timing) of the SRVs in accordance with IWV-3510.

## 1.3 PROPOSED REQUIREMENTS

The set pressure acceptance criteria will be defined as ± 3% by this Technical Specification change. This acceptance criteria coincides to the range defined in ANSI/ASME OM-1 (Reference 4.8), which requires repair or replacement of the valve. The acceptable and expected range for in-service performance is ± 3%. Valves exceeding this criteria are to be repaired or replaced, the cause of the failure determined and corrected, and the valves must successfully pass a retest before being returned to service. Failures of the set pressure acceptance criteria discovered during scheduled Code testing, require an expanded test scope. For each valve of the original test group that fails to meet the acceptance criteria of ± 3%, 2 additional valves are required to be tested in accordance with OM-1. If any of those additional valves fail the acceptance criteria, then all remaining SRVs must be tested. SRVs will be recertified to meet the ± 1% criteria following the as-found testing.

The increased tolerance on the allowable SRV setpoint in Technical Specifications is expected to reduce the number of test failures, and thus the number of LERs and additional valve tests required. This may allow testing a reduced number of valves each outage.

Existing safety/relief setpoints are unaffected by the proposed change. Prior to return of a valve to service, the setpoint will be adjusted to within the current tolerance of 1%, to ensure margin to the ± 3% in-service testing criteria. Following are proposed valve settings for safety/relief mode operation with the ± 3% criteria.

| Number of SRVs | As Found Safety Setting, psig | Recertification Safety Setting, psig |
|---|---|---|
| 5 | 1125 ± 3% | 1125 ± 1% |
| 4 | 1115 ± 3% | 1115 ± 1% |
| 4 | 1105 ± 3% | 1105 ± 1% |

### 1.4 SAFETY ANALYSIS OVERVIEW

The potential safety concerns affected by the SRV safety setpoint tolerance relaxation are discussed in Reference 4.4. They include reactor vessel overpressure protection analyses, ECCS/LOCA performance, fuel thermal limits analyzed for abnormal operational occurrences, containment and SRV discharge loads, and high pressure injection system performance (High Pressure Coolant Injection, Reactor Core Isolation Cooling System, Standby Liquid Control System). A number of these issues were addressed and resolved by the generic licensing topical report (Reference 4.4). The LTR and the NRC SER indicated that certain of these issues, however, must be evaluated on a plant-specific basis. As discussed in Section 2.0, plant-specific analyses have been completed and the proposed change can be made without any impact to plant safety.

## 2.0 <u>BFN PLANT SPECIFIC ANALYSIS</u>

### 2.1 ANALYSIS APPROACH

The NRC SER (Reference 4.5) required that certain issues be evaluated on a plant-specific basis by licensees choosing to implement the Technical Specification change. Plant-specific analyses include analysis of abnormal operational occurrences, design basis overpressurization event, high pressure system performance, motor-operated valves, vessel instrumentation and instrument piping connected to the vessel, plant-specific alternate operating modes, containment response during LOCA, and hydrodynamic loads on the SRV discharge lines and containment. The plant specific analysis for abnormal operational occurrences and the design basis overpressurization event must account for the number of SRVs required to be operable in the Technical Specifications. Each of these analysis areas is discussed in the following sections of this engineering report.

The plant-specific analyses were performed by verifying through bounding calculation or evaluation that existing analyses contain sufficient margin to accommodate the effects of the increase in setpoint tolerance, or by evaluating system or component performance at the higher pressure. Credit was taken for cycle specific reload analysis which was performed using SRV setpoints at the proposed ± 3% increase. The Unit 2 specific analysis has been performed (Reference 4.1) and the Unit 3 specific analysis will be performed prior to implementation on Unit 3.

## 2.2 ANALYSIS OF ABNORMAL OPERATIONAL OCCURRENCES (AOOs)

As stated in GE's Licensing Topical Report NEDC-31753P (Reference 4.4), the effect of SRV opening pressure on the MCPR response must be determined for abnormal operational occurrences (AOOs). The report goes on to state that for Group 1 plants (Target Rock SRVs only, BWR 3/4 class), of which BFN is included, increasing the SRV tolerance to ± 3% will not affect the thermal limits and no plant specific analysis is necessary. Additionally, as required by Reference 4.5, plant-specific analyses have been performed on a cycle-specific basis for Unit 2 at the ± 3% setpoint tolerance for SRVs with one SRV out of service as documented in Reference 4.1 and 4.2. These analyses confirmed the acceptability of the increased SRV (± 3%) setpoint tolerance for all abnormal operational occurrences for Unit 2 Cycle 9. Cycle-specific analyses will be performed for Unit 3 prior to implementation of the ± 3% setpoint tolerance relaxation.

## 2.3 ANALYSIS OF DESIGN BASIS OVERPRESSURIZATION EVENT

GE Licensing Topical Report (Reference 4.4) states that the safety objective of the nuclear pressure relief system is to prevent overpressurization of the nuclear system. To meet this safety objective, it must be demonstrated that the nuclear system stresses are below the applicable industry code limits.

The BFN design basis (worst case) overpressurization event is a closure of all main steam isolation valves while the reactor is operating at 102% rated power and 105% rated core flow. Reactor scram on MSIV position is assumed to fail, so the scram is assumed to occur on high neutron flux. The BFN current analysis only credits 12 of the 13 SRVs (one is assumed inoperable).

The design basis overpressurization analysis for the present Unit 2 fuel cycle (Cycle 9) has been performed. This analysis was performed as part of the normal reload analysis process (Reference 4.1) and determined the maximum vessel pressure to be 1257 psig. This is substantially less than the code allowable limit of 1375 psig.

The Unit 3 overpressurization analysis will be performed using the ± 3% SRV setpoint tolerance prior to implementation on Unit 3. All future reload analyses will be performed to validate the ± 3% SRV setpoint tolerance for both Unit 2 and 3.

## 2.4 HIGH PRESSURE SYSTEMS PERFORMANCE

The High Pressure Core Injection (HPCI), Reactor Core Isolation Cooling (RCIC) and Standby Liquid Control (SLC) systems were considered in this evaluation. The effects of a potential increase in reactor pressure, and thus system operating pressure, due to the increase in SRV safety mode setpoint tolerance from ± 1% to ± 3% is addressed in this section. The HPCI and RCIC systems have a design requirement to provide design flow to the reactor vessel at reactor pressures up to 1120 psig. This is slightly higher than the lowest SRV setpoint (1105 psig), plus the allowable tolerance of ± 1% (11 psig). By increasing the safety mode setpoint tolerance to ± 3%, the pressure at which these systems may potentially be required to operate increases by 22 psi to 1138 psig.

### 2.4.1 HPCI System Evaluation

The HPCI system is an Emergency Core Cooling System (ECCS) which provides a source of high pressure makeup water 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. HPCI also supplies high pressure makeup water during a transient or event that results in the loss of feedwater and/or isolation of the reactor vessel, as required.

#### 2.4.1.1 Pump Capacity Considerations

The HPCI system is designed to deliver a minimum flow of 5000 gpm to the reactor vessel at a pressure ranging from 150 psig to 1120 psig. A review of HPCI pump and turbine governor control system performance data indicated that the HPCI system under its present turbine governor control range can provide the additional head required to support the increased reactor pressure at the ± 3% MSRV setpoint tolerance. From the relationship, $H_1/H_2 = (N_1/N_2)^2$, the HPCI turbine speed would need to increase from 4000 rpm (MD-Q0073-880139) at an injection pressure of 1120 psig to 4032 rpm at an elevated pressure of 1138 psig (1105 psig + 3%) to maintain 5000 gpm flow. This is well below the turbine overspeed trip limit of 5000 rpm and within the current capacity of HPCI as presently configured. The small increase in required rpm also requires an additional time period in order to reach rated speed. This increase in time to rated speed for this small increase in rpm (4000 to 4032) has been evaluated to be insignificant to the overall operation of the HPCI system. Therefore, adequate margin exists for HPCI pump capacity requirements.

#### 2.4.1.2 Turbine Capacity Considerations

The pump total dynamic head increases proportionally with an increase in injection pressure requiring a higher pump speed to produce the increased head. The higher speed and increased pump head require an increase in turbine power and, thus, an increase in steam flow to the turbine. The additional HPCI turbine steam flow requirements were estimated by increasing the pump total dynamic head design value of 2819 ft-$H_20$ (MD-Q2073-890041 and MD-Q3073-92503) at 1120 psig reactor vessel pressure by an amount

equal to the potential increase in maximum vessel injection pressure due to the MSRV setpoint tolerance change (1138 minus 1120), 18 psi or 42 ft-$H_2O$, resulting in a required head of 2861 ft-$H_2O$. The design pump flow rate was not changed.

From the vendor's pump performance curves, at 2819 ft-$H_2O$ head and 4000 rpm, the turbine must deliver 4300 BHP (VTM-B580-0010). As determined above, a speed of 4032 rpm is required for the increased head. From the relationship $BHP_1/BHP_2 = (N_1/N_2)(H_1/H_2)$, the horsepower required at 2861 ft-$H_20$ head and 4032 rpm is 4399 BHP. This is within the 4575 BHP (VTM-G080-6860) rating of the turbine at 4000 rpm.

Turbine steam flow requirements are directly proportional to BHP requirements. At 100% pump flow and 4000 rpm, the turbine requires 130,000 lb/hr steam flow (MD-Q2073-960023). The increased steam flow is: (130,000 lb/hr)(4399 BHP/4300 BHP) = 132,993 lb/hr, which is a 2.3% increase. This is well below the high steam flow trip setting of approximately 408,000 lb/hr (ED-Q2073-880625, 880626 and ED-Q3073-920445). There is a 3 second time delay on the HPCI high steam flow trip which prevents spurious trips during start up transients.

The turbine exhaust header back pressure normally operates in the 30 to 40 psig range. The exhaust line differential pressure will increase with an increase in steam flow according to the relationship $dP_1/dP_2 = (Q_1/Q_2)^2$. Thus, the worst case exhaust back pressure would increase to (40 psig)$(132{,}993 \text{ lb/hr}/130{,}000 \text{ lb/hr})^2$ = 42 psig. The 2 psi increase is inconsequential compared to the 150 psig high back pressure trip setting.

2.4.1.3 Component and Piping Design Considerations

The HPCI pump has a discharge design pressure of 1500 psig. The design pressure of HPCI piping and components from the pump to the feedwater line thermal sleeve piping is 1500 psig and from there to the reactor vessel it is 1375 psig. The design pressure of HPCI steam supply piping connected to the reactor vessel design pressure is 1146 psig (this corresponds to an allowable pressure of 1375 psi per ANSI B31.1) and is protected from overpressure by the MSRVs. These pressures are adequately above the maximum 1138 psig reactor pressure that may result from the MSRV relief setpoint tolerance relaxation.

In conclusion, adequate capacity exists on the HPCI system to accommodate increasing the setpoint tolerance on the MSRVs.

2.4.2 RCIC System Evaluation

The RCIC system shall provide a source of high pressure makeup water to assure that the reactor is adequately cooled so that the Core Standby Cooling Systems are not required. The RCIC also supplies makeup water during the special event of loss of habitability of the control room. The above makeup water function of RCIC is not a safety related function.

2.4.2.1 Pump Capacity Considerations

The RCIC system is designed to deliver a minimum flow of 600 gpm to the reactor vessel at a pressure ranging from 150 psig to 1120 psig and 16 gpm to the lube oil cooler for a total pump discharge flow of 616 gpm. A review of RCIC pump and turbine governor control system performance data indicated that the RCIC system under its present turbine governor control range can provide the additional head required to support the increased reactor pressure at the ± 3% MSRV setpoint tolerance. From the relationship, $H1/H2 = (N1/N2)^2$, the RCIC turbine speed would need to increase from 4438 (VTM-B260-0070) rpm at an injection pressure of 1120 psig to 4474 rpm at an elevated pressure of 1138 psig (1105 psig + 3%) to maintain 616 gpm. This is well below the 125% mechanical turbine overspeed trip limit of 5625 rpm and within the current capacity of RCIC as presently configured. The small increase in required rpm also requires an additional time period in order to reach rated speed. This increase in time to rated speed for this small increase in rpm (4438 to 4474) has been evaluated to be insignificant to the overall operation of the RCIC system. Therefore, adequate margin exists for RCIC pump capacity requirements.

2.4.2.2 Turbine Capacity Considerations

The pump total dynamic head increases proportionally with an increase in injection pressure requiring a higher pump speed to produce the increased head. The higher speed and increased pump head require an increase in turbine power and, thus, an increase in steam flow to the turbine. The additional RCIC turbine steam flow requirements were estimated by increasing the pump total dynamic head design value of 2800 ft-$H_20$ (DC-BFN-50-7071) at 1120 psig reactor vessel pressure by an amount equal to the potential increase in maximum vessel injection pressure due to the MSRV setpoint tolerance change (1138 minus 1120), 18 psi or 42 ft-$H_2O$, resulting in a required head of 2842 ft-$H_2O$. The design pump flow rate was not changed.

From RCIC pump performance curves, at 2800 ft-$H_2O$ head and 4438 rpm, the turbine must deliver 660 BHP (VTM-B260-0070). As determined above, a speed of 4474 rpm is required for the increased head. From the relationship $BHP1/BHP2 = (N1/N2)(H1/H2)$, the horsepower required at 2842 ft-$H_20$ head and 4474 rpm is 675 BHP. This is within the 700 BHP (VTM-6080-6740) rating of the turbine at 4500 rpm.

Turbine steam flow requirements are directly proportional to BHP requirements. At 100% pump flow and 4500 rpm, the turbine requires 38,000 lb/hr steam flow (MD-Q2071-960022). The increased steam flow is: (38,000 lb/hr)(675 BHP/660 BHP) = 38,864 lb/hr, which is a 2.3% increase. This is well below the high steam flow trip setting of approximately 83,000 lb/hr (ED-Q2071-900029 and ED-Q3071-920444). There is a 3 second time delay on the RCIC high steam flow trip which prevents spurious trips during start up transients.

The turbine exhaust header back pressure normally operates in the 15 to 20 psig range. The exhaust line differential pressure will increase with an increase in steam flow according to the relationship $dP1/dP2 = (Q1/Q2)^2$. Thus, the worst case exhaust back pressure would increase to (20 psig)(38,864 lb/hr/38,000 lb/hr)$^2$ = 20.9 psig. The 0.9 psi increase is inconsequential compared to the 50 psig RCIC turbine exhaust pressure trip setpoint. For a cold quick start, the flow rate has been determined to be 47,000 lb/hr with a resulting 29.4 psi back pressure. The small potential increase in back pressure during a cold quick start is inconsequential to the RCI turbine exhaust pressure trip setpoint of 50 psig.

2.4.2.3 Component and Piping Design Considerations

The RCIC pump has a discharge pressure design of 1500 psig. The design pressure of piping and components from the pump to the feedwater line thermal sleeve piping is 1500 psig and from there to the reactor vessel it is 1375 psig. The design pressure of the RCIC steam supply piping connected to the reactor vessel is 1146 psig (this corresponds to an allowable pressure of 1375 psi per ANSI B31.1) and is protected from overpressure by the MSRVs. These pressures are adequately above the maximum 1138 psig reactor pressure that may result from the MSRV relief setpoint tolerance relaxation.

In conclusion, adequate capacity exists on the RCIC system to accommodate increasing the setpoint tolerance on the MSRVs.

2.4.3 Standby Liquid Control System Evaluation

Standby Liquid Control (SLC) system operation is not impacted by the MSRV setpoint tolerance increase because the SLC system pressure parameter is governed by its own system relief valve set at 1425 psig ± 75 psi (TS 4.A.2.a). The SLC system utilizes positive displacement pumps. If the pump discharge pressure is greater than the RPV pressure and piping losses, flow is relatively constant. The minimum limitation of the SLC relief valve setting is intended to prevent the loss of liquid control solution via lifting of a relief valve at too low a pressure. Flow calculations for SLC piping indicate a 162 psi (BWPM2-ATWS-1) piping flow loss at a flow rate of 86 gpm from the pumps to the reactor vessel. With a maximum increased reactor pressure of 1138 psig (1105 psig + 3%) plus SLC piping flow loss (1138 + 162 = 1300 psig) and a minimum 1350 psig (1425 minus 75 psig) SLC relief valve setpoint, a margin of 50 psi exist to prevent loss of SLC injection capability. Therefore, adequate margin exists on the SLC system to accommodate increasing the setpoint tolerance on the MSRVs.

2.5 CONTAINMENT RESPONSE AND HYDRODYNAMIC LOADS EVALUATION

The MSRV setpoint tolerance relaxation to 3% was assessed for potential impact on the containment based on operation with one SRV out of service. The results of the analyses are summarized below.

### 2.5.1 LOCA Containment Responses

The most limiting event in terms of peak containment pressure is the design basis loss of coolant accident (LOCA) consisting of a double ended guillotine break of the recirculation piping (FSAR 14.6.3.3). An increase in the MSRV tolerance has no effect on this event because the vessel depressurizes without any MSRV actuations.

MSRV actuations may occur during small break LOCA scenarios, however, they do not significantly effect the peak pressure since the steam is released to and condensed by the water in the suppression pool. The increased opening pressure will delay the MSRV opening; however, when the SRV opens the flow rate will be higher due to the increased pressure. At these thermodynamic conditions, the 20 psi increase in pressure results in a decrease in enthalpy for the steam released which tends to reduce the impact on containment response. The total mass/energy release will be essentially the same and thus the increase in MSRV tolerance will have a negligible impact on containment pressure response for small break LOCAs requiring MSRV actuations.

The most limiting response for containment temperature is based on two separate events. Initially, the containment design temperature was determined based upon a large break LOCA (i.e., a double ended guillotine break of the recirculation piping). It was later discovered that a small steamline break would eventually produce peak temperatures in excess of the large break. In order to prevent peak containment temperatures following a small break from exceeding the 281°F containment design temperature, the operators can utilize containment spray to cool the containment atmosphere (Reference 4.9). This action would be required to occur prior to 30 minutes following the pipe break to retain the large break results as the limiting event.

For breaks that do not result in MSRV actuations (i.e., large and intermediate size breaks), an increase in the MSRV tolerance has no effect on the containment temperature. For small breaks, the reactor pressure may reach the MSRV setpoint but drop below the MSRV setpoint relatively quickly (Response to original FSAR Question 12.2.16). During this initial period, the containment temperature is still relatively low. The increased MSRV opening pressure will delay the MSRV opening; however, when the MSRV opens the flow rate will be higher due to the increased pressure. At these thermodynamic conditions, the 20 psi increase in pressure results in a decrease in enthalpy for the steam released which tends to reduce the impact on containment response. The total mass/energy release will be essentially the same and thus the increase in MSRV tolerance will have a negligible impact on the long term containment temperature response due to a small steam line break.

### 2.5.2 Safety Relief Valve Dynamic Loads

The dynamic loads due to SRV actuation at BFN have been evaluated to determine the effect of increasing safety open setpoint tolerance from ± 1% to ± 3%. The objective of this evaluation was to verify that the structural margins in the SRV discharge lines, supports, and the wetwell containment vessel are sufficient to withstand increased dynamic loads resulting from higher actuation pressure.

The effects of increased SRV setpoint tolerance have been evaluated in revision one of Reference 4.3. This calculation uses the results of the April 1983 BFN Unit 2 Long Term Torus Integrity Program (LTTIP) post modification SRV tests and the correlation of those test results with analytical predictions to justify conservatism in the SRV loads used for LTTIP analysis and modification designs. The overall effects are assessed relative to compliance with the acceptance criteria in Section 4 and Appendix A of the LTTIP Plant Unique Analysis Report (PUAR), as updated by BFN FSAR Appendix C, Section C3.5 and reflected in the LTTIP design criteria documents. The effects of increased SRV setpoint tolerance have been evaluated for containment structural response from SRV hydrodynamic loads; steam and water clearing loads on SRV lines, quenchers, and supports; increases in hydrodynamic loads on submerged structures; and the effects on piping attached to the torus.

The calculation demonstrates that SRV loads resulting from the 3% setpoint tolerance are less than 1% higher than the loads used in design. When SRV loads are combined with other design basis loads including dead weight, pressure, thermal LOCA, and earthquake, the total load increase is negligible and the affected components meet design basis requirements.

## 2.6 OTHER ANALYSES

### 2.6.1 Generic Letter 89-10 Motor Operated Valve (MOV) Issues

A maximum expected differential pressure (MEDP) was determined for valves in the Generic Letter (GL) 89-10 program to establish a minimum thrust required for valve operation based on expected worst case operational conditions. An MEDP value was established for each MOV in the GL 89-10 Program, and was developed based on industry guidance developed generically for all BWRs, and considering BFN plant specific requirements.

As described in the BWROG LTR (NEDC-31753P) and the Technical Evaluation Report prepared as part of the NRCs evaluation of the LTR, consideration should be given to testing MOVs exposed to reactor pressure at higher differential pressures. For BFN, MOV dynamic testing is done at the highest differential pressure achievable under normal operational configurations for selected valves. Therefore, dynamic testing requirements are unaffected by the SRV setting tolerance increase. However, MOV operator settings for static testing are also based on the MEDP values (as one of the input parameters for determining required settings). Adequacy of MOV settings was assessed by evaluating the basis of the MEDP determinations and the resulting MEDP values established for those valves potentially affected by the increase in SRV safety setpoint tolerance.

The valves which are potentially affected by a change in the SRV setpoint, are those valves which may be expected to perform their safety functions during overpressurization events, and which communicate directly with the Reactor Coolant Pressure Boundary. The valves applicable to BFN and their respective calculations are listed in the table below. A review

of the GL 89-10 calculations performed for these valves shows that an increase of 3% in MEDP results in an increase in the maximum thrust required for valve operation of less than 3%. This review also showed that there is sufficient capability margin in each of the affected MOVs to overcome this increase in MEDP.

| VALVE NUMBER | FUNCTION | CALCULATION |
|---|---|---|
| 2-FCV-01-55 | MS Drain Line PCI Valve | MD-Q2001-910060 |
| 2-FCV-01-56 | MS Drain Line PCI Valve | MD-Q2001-910061 |
| 2-FCV-71-02 | RCIC Steam Line PCI Valve | MD-Q2071-910081 |
| 2-FCV-71-03 | RCIC Steam Line PCI Valve | MD-Q2071-910082 |
| 2-FCV-71-08 | RCIC Steam Admission Valve | MD-Q2071-910083 |
| 2-FCV-73-02 | HPCI Steam Line PCI Valve | MD-Q2073-910093 |
| 2-FCV-73-03 | HPCI Steam Line PCI Valve | MD-Q2073-910094 |
| 2-FCV-73-16 | HPCI Steam Admissions Valve | MD-Q2073-910095 |
| 2-FCV-73-81 | HPCI Steam Line PCI Valve | MD-Q2073-910104 |
| 3-FCV-01-55 | MS Drain Line PCI Valve | MD-3001-920379 |
| 3-FCV-01-56 | MS Drain Line PCI Valve | MD-3001-920380 |
| 3-FCV-71-02 | RCIC Steam Line PCI Valve | MD-3071-920395 |
| 3-FCV-71-03 | RCIC Steam Line PCI Valve | MD-3071-920396 |
| 3-FCV-71-08 | RCIC Steam Admission Valve | MD-3071-920397 |
| 3-FCV-73-02 | HPCI Steam Line PCI Valve | MD-3073-920407 |
| 3-FCV-73-03 | HPCI Steam Line PCI Valve | MD-3073-920408 |
| 3-FCV-73-16 | HPCI Steam Admissions Valve | MD-3073-920409 |
| 3-FCV-73-81 | HPCI Steam Line PCI Valve | MD-3073-920418 |

In summary, MOV settings are based on MEDP values in combination with other parameters. The calculations reviewed show that these valves have sufficient analytical margin. Therefore, the increase in SRV setpoint tolerance to 3% does not adversely affect the ability of the affected valves to perform their safety functions.

2.6.2 Impact on Vessel Instrumentation Piping

Piping connected to the Reactor Coolant Pressure Boundary (RCPB) is designed for pressures equal to or greater than the rated reactor vessel design pressure of 1250 psig. Instrumentation piping and tubing are also designed to meet the process piping pressure rating and therefore any Reactor vessel instrument lines are rated for 1250 psig. The piping, tubing and instrumentation procured and installed were bought to 1500 psig or greater operating pressure. Instrumentation and instrument lines were not reviewed for CRD or SLC systems since their design pressure are higher than the MSRV setpoint tolerances and therefore review of these systems was not necessary.

Therefore, the proposed change in SRV setpoint tolerance has no impact on RCPB piping, including instrument piping and tubing.

### 2.6.3 Instrumentation Evaluation

Instruments which would be affected by the increased pressure resulting from the proposed change were evaluated with respect to effects on pressure boundary integrity, instrument calibration, and instrument setpoint and scaling calculations. Instruments in high pressure systems such as the Control Rod Drive and Standby Liquid Control systems were excluded because the systems are designed to operate at pressures higher than that resulting from the SRV tolerance relaxation and therefore review of these systems was not necessary.

Instrument setpoint and scaling calculations were reviewed for all reactor coolant pressure boundary (RCPB) instruments (ECCS initiation/trip, ATWS Initiation, Containment Isolation, SCRAM initiation, FW Level Control, Shutdown Cooling Reactor Vessel Level instruments). The calculations for these instruments use normal reactor operating pressure or accident (LOCA) pressure for determining water leg errors, static pressure errors, and span error corrections rather than anticipated maximum pressure. These calculations were not affected by the increased anticipated maximum pressure since they are not required for reactor vessel overpressurization or they have performed their trip function prior to reaching the higher pressure The overpressure condition has no adverse effect on the instrumentation after reactor pressure returns to normal operating pressure (e.g., instrumentation is back on scale, no recalibrations required).

The HPCI and RCIC flow calculations were evaluated for the maximum pressure, but there was no impact on the conclusion or instrument settings based on the higher pressure. Calibration frequency is not impacted by proposed change since it is time dependent and not a function of operating pressure.

A review of vendor information for RCPB instruments, indicate that the increased pressure is within the pressure boundary design limit. Instrumentation pressure rating (maximum operating pressure) was 1250 psig or greater. These instruments consists of safety and non safety related, but are required to maintain pressure boundary of the system.

Therefore, the proposed change in SRV setpoint tolerance has no impact on plant instrumentation.

A review of vendor information for RCPB instruments, indicate that the increased pressure is within the pressure boundary design limit. Instrumentation pressure rating (maximum operating pressure) was 1250 psig or greater. These instruments consists of safety and non safety related, but are required to maintain pressure boundary of the system.

Therefore, the proposed change in SRV setpoint tolerance has no impact on plant instrumentation.

### 2.6.4 Anticipated Transients without SCRAM (ATWS)

Section 7.19 of the UFSAR discusses the BFN design to accommodate ATWS. ATWS is considered a low probability event, especially considering the design features BFN has implemented to address 10CFR50.62 to reduce the probability and/or mitigate the consequences of ATWS. The most severe ATWS with respect to peak pressure would be an ATWS coincident with MSIV closure. In an actual ATWS with MSIV closure all

MSIV should lift initially. The important consideration is that they do lift to relieve RPV pressure. The potential increase in lift pressure (22 psig) due to the setpoint tolerance change is insignificant to the overall successful mitigation of an ATWS event. A generic BWR analysis (Reference 4.10) was prepared to investigate transient response to ATWS events. This analysis was reviewed and it was concluded that the results were not significantly affected by this setpoint tolerance change. Therefore, the ± 3% MSRV setpoint tolerance allowance does not affect BFN capability to mitigate an ATWS event.

### 2.7 EVALUATION OF PLANT SPECIFIC OPERATING MODES

The Unit 2 Cycle 9 reload analysis (Reference 4.1) was performed for an MSRV setpoint tolerance of ± 3%. This analysis bounds the current BFN Unit 2 licensed operating power/flow map. The Extended Load Line Limit and Increased Core Flow as well as the Final Feedwater Temperature Reduction flexibility option are analyzed assuming an MSRV setpoint tolerance of ± 3%. The Unit 3 specific reload analyses will be performed prior to implementation for Unit 3.

## 3.0 CONCLUSION

The analyses summarized above support the relaxation of the tolerances for MSRV setpoint to ± 3% from the current ± 1%.

## 4.0 REFERENCES

4.1 General Electric Company, Supplemental Reload Licensing Report for Browns Ferry Nuclear Plant, U2, Reload 8, Cycle 9, J11-02761 SRLR, Rev. 3, May 1996

4.2 General Electric Company, MSRV Allowable Setpoint Tolerance, BFSE 96-060, October 31, 1996

4.3 Calculation CD-Q0999-960064, Rev. 1, Impact of Increased Torus Water Level on BFN Long Term Torus Integrity Program (LTTIP), R14 961101 101

4.4 K. F. Conwell, et. al., "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report", General Electric Company, February 1990 (NEDC-31753P)

4.5 Letter, Ashok C. Thadani (NRC) to Cynthia L. Tully (BWROG), "Acceptance for Referencing of Licensing Topical Report NEDC-31753P, 'BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report'" (NRC Safety Evaluation Report Enclosed), March 8, 1993

4.6 BFN Updated Final Safety Analysis Report, Amendment 13

4.7 BFN Technical Specifications, Through Amendment 231 (U1), 246 (U2), 206 (U3)

4.8 ANSI/ASME OM-1-1981, Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices (as referenced in Subsection IWV-3500 of the ASME Code, Section XI, 1986 Edition)

4.9 General Electric Company, Impact of Increased Containment Pressure and Temperature for Browns Ferry Units 1, 2, and 3, GE-NE-T2300733-00, Supplement 1, April 1, 1996

4.10 General Electric Company, Assessment of BWR Mitigation of ATWS, NEDE-24222 (GE Proprietary), December 1979