GRAND GULF NUCLEAR STATION
ENGINEERING REPORT

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

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EXECUTIVE SUMMARY

This report presents the results of the Grand Gulf Nuclear Station (GGNS) plant-specific evaluations (References 4.1, 4.2, 4.3 & 4.9) to support increasing the safety/relief valves (SRV) safety mode setpoint tolerance from ± 1% to ± 3%. The current requirements for the SRV relief mode, Low-Low Set mode and 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 area of 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, 4.3 and 4.9, 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 GGNS 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. GGNS 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, 4.3 & 4.9) required to support the change.

The vessel overpressure protection system at GGNS 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 20) 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 spray (HPCS) system so that the low pressure coolant injection (LPCI) and the low pressure core spray (LPCS) systems can operate to protect the fuel barrier.

GGNS utilizes DIKKERS 8"x10" dual function safety/relief valves which open in a direct acting safety mode in response to system pressure acting on the valve's spring loaded disk, and open in the relief mode using an auxiliary actuating device consisting of a pneumatic piston/cylinder and mechanical linkage assembly which opens the valve by overcoming the valve spring force.

1.2 PRESENT REQUIREMENTS

GGNS Technical Specifications (Reference 4.7) require that the safety function of seven SRVs be operable, and the relief function of six additional SRVs be operable. The nominal opening setpoints and tolerances for the valves in each mode are as follows:

Number of SRVs	Safety Setting, psig	Relief Setting, psig
8	1165 ± 11.6	1123 ± 15
6	1180 ± 11.8	1113 ± 15
4	1190 ± 11.9	1113 ± 15
1	1190 ± 11.9	1123 ± 15
1	1190 ± 11.9	1103 ± 15

Current setpoint tolerance is based on ASME Code requirements for pressure relief devices operating in the pressure range described above. Operability of the valves is based on Technical Specification surveillance test acceptance criteria for an as-found setpoint tolerance of 1%. Exceeding the 1% tolerance requires performance of evaluations to determine if the plant has operated in an unanalyzed condition, and Licensee Event Report (LER) initiation. In addition, test failures based on the 1% criteria would result in testing additional valves in accordance with Code requirements. (GGNS has, to this point, replaced all twenty SRVs each outage to preclude outage schedule impact due to additional valve testing requirements as a result of failure to meet the 1% criteria.)

The GGNS In-Service Testing (IST) Program is currently performed in accordance with the requirements of the ASME Code, Section XI, 1980 Edition with Addenda through Winter 1980. The SRVs at GGNS are classified as both Category B and Category C valves according to the GGNS IST Program. Category B test requirements for the SRVs are addressed in IST Program Relief Request B21-3, which is not affected by the proposed change in SRV safety mode setpoint tolerance. The alternative testing required by Relief Request B21-3, in lieu of Code specified Category B valve testing, is to perform relief mode operability testing of the valves during plant start-up after each refueling as required by the Technical Specifications. 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-3500.

1.3 PROPOSED REQUIREMENTS

The ASME has revised the setpoint tolerance acceptance criteria, for demonstrating operational readiness of the SRVs, from 1% to 3% per Reference 4.8. The acceptance criteria defines the range of expected in-service performance of a valve. 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. In addition, for each valve that fails to meet the acceptance criteria of 3%, two additional valves are to be tested, as required by ASME OM-1 and the NRC SER.

The increased tolerance on the test acceptance criteria to ±3% for IST and for operability determination per Tech Specs is expected to reduce the number of test failures, and thus the number of LERs and additional valve tests required. This will allow testing a reduced number of valves each outage without significant risk of outage impact.

Existing safety mode and relief mode 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 mode operation with the 3% criteria.

Number of SRVs	As Found Safety Setting, psig	Recertification Safety Setting, psig
8	1165 ± 35	1165 ± 11.6
6	1180 ± 35.4	1180 ± 11.8
6	1190 ± 35.7	1190 ± 11.9

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 Core Spray, 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 GGNS PLANT SPECIFIC ANALYSIS

2.1 ANALYSIS APPROACH

The NRC SER (Reference 4.5) indicated 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 8 reload analysis which was performed using a SRV setpoint tolerance in excess of the proposed 3% increase.

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 BWR 6's (i.e., GGNS), credit may be taken for externally powered valves. The generic conclusion in the GE report states that BWR 6's (i.e., GGNS) need not consider relaxation of the safety/relief valves safety function lift setpoint tolerances for its affect on the MCPR response for AOOs since the relaxation of setpoint tolerances apply only to the safety mode of the safety/relief valves and do not affect the relief mode.

GGNS has dual-mode (relief and safety) SRVs and takes credit for the externally powered relief mode of SRV actuation when evaluating AOOs. However, GGNS only takes credit for six valves in the relief mode while taking credit for seven valves in the safety mode. The relief mode setpoints are not affected by a relaxation of the safety/relief valves safety function lift setpoint tolerances. Because the safety mode of seven safety/relief valves are taken credit for, each AOO considered in the current reload analysis (Safety Evaluation 95-0022-R00, Reference 4.9) is discussed below.

2.2.1 Load Reject No Bypass (LRNB)

UFSAR Section 15.2.2, presents the analysis for generator load rejection. UFSAR Subsection 15.2.2.2 discusses sequence of events and system operation for this AOO. UFSAR Subsection 15.2.2.2.2.1 states: "The pressure relief system which operates the relief valves independently when system pressure exceeds relief valve instrumentation set points is assumed to function normally during the time period analyzed." UFSAR Subsection 15.2.2.2.2, Generator Load Rejection with Failure of Bypass, states: "Six safety/relief valves in the relief mode and seven in the safety mode are assumed to be available. The opening setpoints used in the analyses and other significant input parameters and initial conditions are listed in Table 15.0-4." UFSAR Table 15.0-4 lists the safety valves number and opening setpoints as follows:

Number of S/RVs	Setpoint (psia)	(psig)
1	1251	1236.3
4	1267	1252.3
2	1277	1262.3

GGNS Technical Specifications Surveillance Requirement SR 3.4.4.1 states: "Verify the safety function lift setpoints of the required S/RVs are as follows:"

Number of S/RVs	Setpoint (psig)
8	1165 ± 11.6
6	1180 ± 11.8
6	1190 ± 11.9

Converting the above Tech Spec setpoints to a ± 3% tolerance instead of the ± 1% tolerance shown:

Number of S/RVs	Setpoint (psig)
8	1165 ± 35.0 (i.e., 1200.0 psig, max)
6	1180 ± 35.4 (i.e., 1215.4 psig, max)
6	1190 ± 35.7 (i.e., 1225.7 psig, max)

As can be seen above, the lowest safety/relief valve safety function setpoint used in the analysis of UFSAR Section 15.2.2 (i.e., 1251 psia or 1236.3 psig) would be at least 10 psig higher than the highest Tech Spec allowable value if a 3% tolerance is used (i.e., 1225.7 psig). Therefore, the current analysis bounds the increase in safety function setpoint tolerance to ± 3%.

Additionally, as stated in the GE Licensing Topical Report (Reference 4.4), the MCPR for this event typically occurs prior to the actuation of any safety/relief valve. Because of this, increasing the safety/relief valve safety function setpoint would not affect the MCPR during this event. The GE Licensing Topical Report also considered the effect of the (-)3% tolerance. It is anticipated that the lowering of the safety function setpoint would make the event less severe because the lowered setpoint could cause the safety/relief valves to open prior to the time of MCPR. It is expected that the MCPR would remain unchanged or improve since the earlier SRV actuation would reduce the vessel pressurization.

2.2.2 Feedwater Controller Failure (Upscale) No Bypass (FWCFNB)

UFSAR Section 15.1.2, presents the analysis for feedwater controller failure - maximum demand. UFSAR Subsection 15.1.2.2 discusses sequence of events and system operation for this AOO. UFSAR Subsection 15.1.2.3.2 states: "The analyses have been performed, unless otherwise noted, with the plant conditions identified in Section 15.0 for the reload transients (Table 15.0-4)." and "The safety/relief valve action is conservatively assumed to occur with higher than normal set points." UFSAR Table 15.0-4 assumes six safety/relief valves in the relief mode and seven in the safety mode are available.

See the discussion above for the LRNB event concerning existing analysis assumptions as they relate to the proposed safety/relief valve safety function setpoints tolerance relaxation. Based on the LRNB discussion, the lowest safety/relief valve safety function setpoint used in the analysis of UFSAR Section 15.1.2 (i.e., 1251 psia or 1236.3 psig) would be at least 10 psig higher than the highest Tech Spec allowable value if a 3% tolerance is used (i.e., 1225.7 psig). UFSAR Subsection 15.1.2.3.3, Results, states: "The ΔCPRs calculated for this event are less than or equal to the LRNB results and support the established MCPR operating limits." Therefore, the current analysis bounds the increase in safety function setpoint tolerance to ± 3%.

Additionally, as stated in the GE Licensing Topical Report (Reference 4.4), the MCPR for this event typically occurs prior to the actuation of any safety/relief valve. Because of this, increasing the safety/relief valve safety function setpoint would not affect the MCPR during this event. The GE Licensing Topical Report also considered the effect of the

(-)3% tolerance. It is anticipated that the lowering of the safety function setpoint would make the event less severe because the lowered setpoint could cause the safety/relief valves to open prior to the time of MCPR. It is expected that the MCPR would remain unchanged or improve since the earlier SRV actuation would reduce the vessel pressurization.

2.2.3 Control Rod Withdrawal Error (RWE)

UFSAR Section 15.4.2, presents the analysis for rod withdrawal error (RWE) at power. UFSAR Subsection 15.4.2.2 discusses sequence of events and systems operation for this AOO. UFSAR Subsection 15.4.2.4, Barrier Performance, states: "Typically, an increase in total core power for RWEs initiated from rated conditions is less than 4 percent and the changes in pressure are negligible." The analysis does not anticipate actuation of any safety/relief valves for this AOO (reference UFSAR Table 15.4-1). This is supported by the GE Licensing Topical Report (Reference 4.4). Therefore, the current analysis bounds the increase in setpoint tolerance to ± 3% since the analysis is not dependent on safety/relief valve safety function setpoint.

2.2.4 Recirculation Flow Control Failure

UFSAR Section 15.4.5, presents the analysis for recirculation flow control failure with increasing flow. UFSAR Subsection 15.4.5.2 discusses sequence of events and systems operation for this AOO. UFSAR Subsection 15.4.5.4.1, Barrier Performance Fast Opening of One Recirculation Valve, states: "This transient results in a very slight increase in reactor vessel pressure as shown in Figure 15.4-4 and therefore represents no threat to the RCPB." UFSAR Subsection 15.4.5.4.2, Barrier Performance Fast Opening of Two Recirculation Valves, states: "This transient results in a very slight increase in reactor vessel pressure as shown in Figure 15.4-5 and therefore represents no threat to the RCPB." UFSAR Subsection 15.4.5.4.3, Barrier Performance Slow Opening of a Recirculation Flow Control Valve, states: "This event results in a final pressure corresponding to the final steady state power. Because of the quasi steady state nature of the event, there is no threat to the reactor coolant pressure boundary." The analysis does not anticipate actuation of any safety/relief valves for this AOO (reference UFSAR Figures 15.4-4 and 15.4-5). This is supported by the GE Licensing Topical Report (Reference 4.4). Therefore, the current analysis bounds the increase in setpoint tolerance to ± 3% since the analysis is not dependent on safety/relief valve safety function setpoint.

2.2.5 Loss of Feedwater Heating (LOFWH)

UFSAR Section 15.1.1, presents the analysis for loss of feedwater heating. UFSAR Subsection 15.1.1.2 discusses sequence of events and systems operation for this AOO. UFSAR Subsection 15.1.1.3 states: "The response to the LOFWH transient is relatively slow, with the reactor core remaining in a nearly steady state condition throughout the event." UFSAR Subsection 15.4.5.4.1, Barrier Performance, states: "The consequences of this event do not result in any pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed....." UFSAR Subsection 15.1.1.5, Radiological Consequences, states: "Radiological consequences were not evaluated since no fuel failures are associated with the event and no radioactivity is discharged to the

suppression pool." The analysis does not anticipate actuation of any safety/relief valves for this AOO. This is supported by the GE Licensing Topical Report (Reference 4.4). Therefore, the current analysis bounds the increase in setpoint tolerance to ± 3% since the analysis is not dependent on safety/relief valve safety function setpoint.

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 GGNS design basis (worst case) overpressurization event is a closure of all main steam isolation valves while the reactor is operating at 100% 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 BWR 6 design meets the ASME Section III Article NB 7542 allowance that up to half of the installed S/RVs may take credit for the auxiliary actuating device (relief mode); however, it should be noted that the GGNS analysis only credits six of the installed twenty S/RVs for actuation in the relief mode.

The design basis overpressurization analysis for the upcoming fuel cycle (Cycle 9) has not been performed. This analysis will be performed as part of the normal reload analysis process. For cycle 9, this analysis will be performed by GE using the NRC approved ODYN methodology described in GESTAR-II, NEDE-24011-P-A.

Overpressurization analyses performed for previous fuel cycles using SRV opening pressures with tolerances in excess of the ± 3% showed considerable margin (≈100 psi) to the ASME Boiler and Pressure Vessel Code limit of 1375 psig. This margin also ensures that the GGNS Operating License Reactor Coolant System Pressure Safety Limit of 1325 psig is not exceeded. Based on these results and the relative insensitivity of the results to the fuel design parameters, future analyses including that planned for Cycle 9, are expected to yield peak pressures with similar margin to ASME Boiler and Pressure Vessel Code limit. As required by the SER, future reload safety analyses will bound the proposed ± 3% safety setpoint tolerance.

2.4 HIGH PRESSURE SYSTEMS PERFORMANCE

The High Pressure Core Spray (HPCS), Reactor Core Isolation Cooling (RCIC) and Standby Liquid Control (SLC) systems were considered in this evaluation. Reference 4.3 addressed 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%. The HPCS and RCIC systems have a design requirement to provide design flow to the reactor vessel at reactor pressures up to the lowest SRV safety setpoint of 1165 psig, plus the allowable ASME Code tolerance of 1% (11.6 psig), or 1177 psig. By increasing the safety mode setpoint tolerance to ± 3%, and considering that not all valves will be tested and reset each outage, the pressure at which these systems may potentially be required to operate increases by approximately 23 psi, to 1200 psig.

2.4.1 HPCS System Evaluation

The HPCS system is an Emergency Core Cooling System (ECCS) which is designed to deliver sufficient coolant to the reactor core, in conjunction with other ECCS systems, to prevent excessive fuel cladding temperature in the event of a loss-of-coolant accident (LOCA) through any design basis break of the reactor coolant boundary. The HPCS also supplies makeup water to the reactor vessel in the event of a transient which results in the loss of feedwater flow and/or isolation of the reactor, with failure of the RCIC system.

The HPCS system is designed to deliver a minimum water flow of 550 gpm to the reactor vessel with reactor vessel pressure 1177 psi above the pressure at the source of pump suction. A review of HPCS vendor and preoperational test pump performance data indicated that the HPCS system has sufficient margin to deliver flow in excess of the 550 gpm minimum at the elevated vessel pressure of 1200 psig (with pump suction aligned to either the Condensate Storage Tank or to the Suppression Pool). Recent GGNS calculations to support system modifications further support this conclusion.

The HPCS pump design has a discharge pressure rating of 1575 psig. The design pressure rating of HPCS discharge piping from the pump to the reactor vessel is 1575 psig. These pressures are well above the pressures which may result from the SRV safety mode setpoint tolerance relaxation; therefore, adequate margin exists.

Additionally, the vendor pump performance data indicated that the slightly lower flow rate expected at the increased discharge pressure results in a slight decrease in electrical load supply requirements for the HPCS Diesel Generator.

2.4.2 Reactor Core Isolation Cooling System Evaluation

The RCIC system is designed to assure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling to take place during the following conditions:

- Should the vessel be isolated and maintained in the hot standby condition.
- Should the vessel be isolated and accompanied by loss of coolant flow from the reactor feedwater system.
- Should a complete plant shutdown, under conditions of a loss of normal feedwater flow, be started before the reactor is depressurized to a level where the Residual Heat Removal (RHR) System shutdown cooling mode can be placed into operation.
- Used in conjunction with HPCS, should a design basis control rod drop accident occur.

Additionally, the RCIC system may be used to provide reactor vessel makeup for protection against small breaks/leaks in the reactor coolant pressure boundary so that fuel design limits are not exceeded.

The RCIC system is designed to deliver a flow of 800 gpm to the reactor vessel at a reactor operating pressure range of 150 psig to 1177 (1165 + 1%) psig within 30 seconds following conditions which cause a system initiation.

The pump total dynamic head increases proportionally to an increase in injection pressure, requiring a higher pump speed to produce the increased head. The higher speed and increased pump head requires an increase in turbine power, and steam flow to the turbine.

The RCIC turbine operational requirements were estimated at the increased reactor pressure due to the increased SRV safety setpoint tolerance (1165 + 3%), by increasing the pump total dynamic head design value of 2980 ft by an amount corresponding to the potential increase in maximum discharge pressure (23 psi, or approximately 55 ft), resulting in a required head of 3035 ft. The required pump flow rate was not changed. Pump vendor test data and pump affinity laws were used to determine the corresponding turbine speed and horsepower requirements; and the required steam flow rate was calculated based on a proportional increase with the increase in turbine horsepower.

Based on the vendor turbine performance curves, the turbine has the capability of producing the required horsepower at the increased pumping head. At the higher steam flow rate and turbine speed, the turbine is capable of producing approximately 1000 horsepower, which is well above that required (slightly less than 900 bhp) for the increase in pump head. Therefore, the ability of the RCIC unit to meet the increased power and speed requirements is within the vendor demonstrated turbine operating and performance limits.

The RCIC system is designed to deliver rated flow to the reactor vessel within 30 seconds following a system initiation. The additional time required for the turbine to reach the higher speed is relatively small (an additional 0.20 sec), and is insignificant to the overall requirement of 30 seconds. RCIC unit startup tests indicate that there is a 6(+) sec margin in the time required to reach rated flow; therefore, the system's ability to meet the startup performance requirements is not impacted.

With the increased steam pressure at the turbine, initial acceleration rate will increase, making governor control more critical. Grand Gulf has installed a turbine startup bypass modification, and with this modification the ability of the turbine control system to prevent an overspeed trip during system startup was greatly enhanced. Based on the startup test data, which was obtained post-modification, and data obtained during subsequent plant transients, turbine governor response provided adequate margin to overspeed during start transients and, therefore, margin to overspeed during system startup at the increased reactor pressure is not adversely affected.

Turbine steam flow rate is expected to increase by approximately 2.7%. The resultant increase in turbine exhaust pressure will be negligible and is expected to have no adverse affect on turbine, or associated equipment performance. The turbine exhaust pressure is expected to remain less than 10 psi (pressures of approximately 9 psi were observed during startup testing) which is well below the turbine high exhaust pressure trip setpoint (25 psig), and burst pressure of the rupture diaphragms (150 psig).

RCIC steam flow is monitored by flow instrumentation for the purpose of isolating steam to the turbine if an excessive steam flow rate, indicative of a line break, occurs. The instrumentation setpoint is equivalent to 300% of steam flow at the design condition of 1192 psia and 800 gpm pump flow. Although operation beyond this design condition requires slightly increased steam flow, the margin to isolation on high steam flow is not significantly impacted. The existing high steam flow setpoint of 56 inches of water, is

slightly conservative in comparison to a setpoint based on 300% of steady state steam flow at 1215 psia.

The RCIC pump design discharge pressure rating is 1525 psig. RCIC system discharge piping has a design pressure rating of 1500 psig. These pressures are above the pressures which result from the SRV setpoint tolerance relaxation; therefore, adequate margin is retained. Steam inlet piping to the RCIC turbine is rated the same as the reactor pressure vessel design of 1250 psig, and with the same overpressure protection requirements; therefore, adequate margin exists (see Section 2.3).

Based on a review of the transient and accident analyses in the GGNS UFSAR, the likelihood of RCIC being required to start and/or inject with reactor pressure at 1200 psig (or even 1177 psig) is extremely remote. This is true since the pressure transient in the vessel is typically being controlled by SRVs and is decreasing when RCIC is signaled to start, and (pressure) is under control of the low-low set relief valve(s) when RCIC has completed its startup sequence and is ready to inject.

2.4.3 Standby Liquid Control System Evaluation

SLC system operation is not impacted by the SRV safety setpoint tolerance increase because the reactor pressure used to analyze system performance is based on SRV relief mode settings, rather than on safety settings. SRV relief mode settings and associated tolerances are not affected by this change.

2.5 CONTAINMENT RESPONSE AND HYDRODYNAMIC LOADS EVALUATION

The SRV safety mode setpoint tolerance relaxation to 3% was assessed for potential impact on the containment hydrodynamic loads at GGNS in General Electric Company, Containment Loads and ECCS-LOCA Analyses for Grand Gulf Nuclear Station Safety/Relief Valve Tolerance Relaxation (Reference 4.1). The analyses consider operation with up to seven SRVs out of service. The results of the analyses are summarized below.

2.5.1 LOCA Containment Response

The design basis (DBA) loss of coolant accident (LOCA), a double-ended guillotine break of the steam line, is the most limiting event in terms of peak containment pressure and temperature and peak suppression pool temperature for GGNS. During this event, the vessel depressurizes through the steam line break and thus there are no SRV actuations. Therefore, the relaxation of the SRV safety mode setpoint tolerance has no affect on DBA-LOCA peak containment pressure and temperature or peak suppression pool temperature.

Small steam line breaks can require SRV actuations to depressurize the vessel. However, an increase in the SRV safety mode setpoint tolerance, which does not affect low-low setpoint tolerances, will have a negligible effect on peak drywell temperature and peak suppression pool temperature, and will not impact the containment pressure and temperature response for small steam line breaks.

The effect of any containment pressure and temperature changes on the LOCA hydrodynamic loads such as pool swell, condensation oscillation, and chugging were evaluated as described in Reference 4.1. The LOCA hydrodynamic loads were found to be unaffected.

2.5.2 Safety Relief Valve Dynamic Loads - From SRV To First Anchor Point and Quencher

The SRV dynamic loads defined for GGNS were reviewed to determine the effect of a relaxation of the SRV safety open setpoint tolerance to ± 3%. The purpose of the review was to determine if sufficient conservatism and margins in the GGNS defined SRV loads are available to offset the effects of an increase in the SRV opening pressure of 3%.

The SRV loads resulting from SRV operation include the reaction and thrust loads acting on the SRV discharge line (SRVDL) and quencher and the air-bubble loads which are transmitted to the submerged boundaries and structures. These loads and the basis for these loads as applied to GGNS are summarized in Appendix 6A of the GGNS UFSAR. An increase in the SRV safety mode open setpoint tolerance to 3% from the current value of 1% will result in an increase in the SRV opening discharge flow rate into the SRV discharge line. This in turn results in an increase in the loads associated with SRV openings. An evaluation of the impact on the SRV loads was performed as described in Reference 4.1. Based on the results of the evaluation, it is concluded that with existing conservatisms in the SRVDL loads a relaxation of the safety setpoint pressure to ±3% will not result in any allowable stresses being exceeded in the SRVDL piping and supports between the SRV and the first anchor point.

GGNS uses the standard GE X-quencher. The loads defined for the quencher were evaluated as described in Reference 4.1 and are not affected by the relaxation of the SRV safety mode open setpoint to ±3%.

2.5.3 Safety/Relief Valve Discharge Lines - First Anchor Point to Quencher (AE Scope Piping)

A calculation (Reference 4.2) was done to determine the effects of the increased pressure resulting from the SRV setpoint tolerance relaxation on the SRV discharge piping from the first anchor point on the discharge piping. The SRV discharge flow rate and resultant hydrodynamic loads on the discharge increases in proportion to the setpoint pressure. The initial calculations to determine discharge pipe loads were done using a valve setpoint pressure greater than the actual setpoint pressures, providing conservatism to the existing analyses. For this evaluation, valve setpoint pressures with a +3% tolerance were compared to the setpoint pressure used in the existing analyses. The calculation indicated that for 14 of the 20 SRVs, increasing the setpoint tolerance from 1% to 3% has no effect on the existing analyses. The remaining six valves experience approximately 0.7% increase in thrust loads due to the setpoint tolerance increase. The thrust loads are combined with deadweight, thermal expansion and other DBA loads in the structural analysis of the piping, supports and quencher. Therefore, the net effect due to the setpoint tolerance increase is negligible.

2.5.4 Submerged Pool Boundary and Structural Loads

Loads on the submerged boundary and on submerged structures are based on the peak bubble pressure determined with the generic GE X-quencher methodology. The conservatisms in the generic GE X-quencher methods were reviewed to address load increases due to the SRV safety mode setpoint tolerance relaxation. Based upon the results of the evaluation described in Reference 4.1, there is no affect on the GGNS SRV pool boundary load definition and loads currently defined for the GGNS submerged structures are also not affected by the relaxed setpoint tolerance.

2.5.5 Effect of Operation with Up to Seven SRVs Out-of-Service

The effect of operation with up to seven SRVs out of service was assessed as described in Reference 4.1. There is no impact of seven SRVs out-of-service on the symmetric and asymmetric SRV loads at GGNS.

2.6 OTHER ANALYSES

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

A maximum expected differential pressure (MEDP) calculation is performed for valves in the Generic Letter 89-10 program to establish a maximum differential pressure based on expected worst case operational conditions. MEDP calculations were prepared for MOVs included in the GL 89-10 Program, and were developed based on industry guidance developed generically for all BWRs, and considering GGNS plant specific requirements.

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

Plants defined in the LTR as Group 3 (i.e., BWR 5/6), have dual mode of operation (relief and safety) SRVs and credit the safety-grade externally powered relief mode in the analyses of abnormal operational occurrences (AOOs), as well as the ASME overpressurization analysis. For GGNS, transient and accident analyses (with exception of the Low-Low Set Relief Function analysis) in the SAR credit the Technical Specifications complement of SRVs. This includes seven SRVs in the safety mode and six SRVs in the externally powered relief mode. For the SRVs operating in the relief mode there are three groups of relief setpoints. The highest setpoint group opens at 1123 psig ±15 psig. Thus, the highest relief mode pressure is 1138 psig, and all SRVs credited in the accident and transient analyses for relief mode operation would open if reactor pressure exceeded this value. Given that this is approximately 40 psi below the lowest SRV safety setpoint of 1165 psig plus 1% tolerance, the maximum differential pressure for those MOVs listed

above remains bounded by the calculated MEDP values that are based on the safety setpoint of the SRVs. Therefore, the use of the reactor pressure value based on the existing SRV safety mode setpoint plus the 1% tolerance as the basis for the calculated MEDP remains a valid assumption and is unaffected by the SRV safety setpoint tolerance increase.

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 rated reactor vessel design pressure of 1250 psig. Instrument piping/tubing class is determined by the process pipe class. Additionally, instrument piping connected directly to the reactor vessel has a minimum design pressure rating of 1250 psig. All of this piping is protected from overpressurization by the SRVs which satisfy ASME Code requirements for overpressure protection for the reactor vessel and connected piping. Pressure transients associated with upset and faulted conditions analyzed in the UFSAR are bounded by core reload analyses which utilize a +6% tolerance for SRV safety mode operation in evaluating maximum overpressurization scenarios (Reference 4.9). Therefore, RCPB piping, including the instrument piping within the RCPB, has adequate design margin for overpressure protection.

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 scaling and setpoint/uncertainty calculations (as applicable). 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.

A review of vendor information for each instrument indicated that the increased pressure is within the pressure boundary design limit. Calibration information for the instruments was reviewed and the calibration range of all instruments, whose range is inclusive of 1177 psig, is adequate considering a potential higher reactor pressure of 1200 psig.

Instrument scaling calculations use normal operating pressures as an input, rather than anticipated maximum pressures and are, therefore, not affected by this proposed change. Additionally, the reactor pressure values used in determination of static pressure effects and overpressure effects in instrument setpoint and uncertainty calculations bound the pressure resulting from the proposed SRV setpoint tolerance relaxation.

Therefore, the proposed change in SRV setpoint drift tolerance has no impact on plant instrumentation and instrument piping/tubing.

2.6.4 Anticipated Transients without SCRAM (ATWS)

Section 15.8 of the UFSAR discusses the GGNS design to accommodate ATWS. ATWS is considered a low probability event, especially considering the design features GGNS has implemented to address 10CFR 50.62 to reduce the probability and/or mitigate the consequences of ATWS. Therefore, ATWS is not analyzed as an Abnormal Operational Occurrence. ATWS has been analyzed generically for BWRs as described in UFSAR Section 15.8. Reference 4.10 provides the results of transient analysis conducted by General Electric to document the response of the BWR to proposed ATWS mitigation systems. This analysis assumed that the SRVs open at their relief pressure setpoint. Therefore, relaxation of the SRV safety function setpoint tolerance will have no effect on the results of ATWS analysis.

2.7 EVALUATION OF PLANT-SPECIFIC ALTERNATE OPERATING MODES

2.7.1 Maximum Extended Operating Domain

All transients and accident analyses performed in support of the Cycle 8 reload incorporated the extended load line and increased core flow available under the GGNS MEOD. For example, rated analyses have been performed at 104.2% rated power and 108% rated flow in order to bound the 100% power, 105% flow combination possible under MEOD.

2.7.2 Single Loop Operation (SLO)

GGNS operation with a single recirculation loop out of service for a period of time has been analyzed by GE. This analysis included reviews and re-analysis of the applicable transients. These analyses concluded that since the plant cannot reach rated bundle powers in SLO due to the reduced core flow capabilities of a single loop, the three most limiting transients are (per UFSAR Section 15C.3.1): feedwater controller failure (maximum demand), load reject no bypass, and pump seizure.

As discussed in UFSAR Section 15C.3.1 for cycle 6, as a result of GE's analyses, Siemens Power Corporation has determined that the pump seizure accident is a limiting event at GGNS during SLO. The results of this analyses indicate that no parameters will be exceeded as a result of the proposed change to the safety function setpoint tolerance.

2.7.3 Feedwater Heaters Out of Service

GGNS is licensed to operate with FWHOS with feedwater temperatures as low as 370° F. AS discussed in UFSAR Section 15.1.2.3.3, the feedwater controller failure (FWCF) with FWHOS was found to be bounded by the FWCF event without bypass. This conclusion was restated in the Cycle 8 fuel reload safety analysis (Reference 4.9). The FWHOS operational mode is therefore bounded by previously analyzed Abnormal Operational Occurrence described in Section 2.2.2.

3.0 CONCLUSIONS

The analyses summarized above support the relaxation of the tolerances for the SRV safety mode open setpoints to ± 3% from the current ± 1%.

4.0 REFERENCES

4.1 General Electric Company, Safety Review for Grand Gulf Nuclear Station Safety/Relief Valves Setpoint Tolerance Relaxation/Out-of-Service Analyses, GE-NE-B21-00599-0296 (GE Proprietary), February 1996

4.2 Engineering Calculation, 8.8.15Q Supplement 1, Revision 0, Main Steam Safety/Relief Valve Discharge Vent Force Transients

4.3 Engineering Report, GGNS-95-0055, Safety/Relief Valve Setpoint Tolerance Relaxation, Revision 0

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

4.5 Letter, Asnok 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 GGNS Updated Final Safety Analysis Report, Revision 9

4.7 GGNS Technical Specifications, through Amendment 122

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 GIN-95/01458, Nuclear Plant Engineering Issuance of 10CFR50.59 Safety Evaluation No. 95-0022-R00, Cycle 8 Reload Analysis, May 3, 1995

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