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LIMERICK GENERATING STATION UNITS 1 AND 2 SRV SETPOINT TOLERANCE RELAXATION LICENSING REPORT

Y. C. Chu
Nuclear & Safety Analysis

R. T. Earle 1/7/99
R. T. Earle
Systems Engineering

Approved by: K. R. Fletcher, Manager
Nuclear & Safety Analysis

Approved by: FOR
P. T. Tran, Project Manager
Engineering & Licensing Projects

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SUMMARY

This report provides the technical justification to support the relaxation of the safety/relief valve (SRV) opening setpoint tolerances for the Limerick Generating Station (LGS) Units 1 and 2 from the current value ±1% to a new value of ±3%. The technical justification addresses the safety-related issues associated with the proposed changes in the following areas:

- ***Vessel overpressurization*** – It is determined that only two SRVs are allowed to be out-of-service with ±3% setpoint tolerance relaxation to meet both overpressure criteria.

- ***Fuel thermal limits during anticipated operational occurrences*** – Results show that there is no impact on the fuel thermal limits due to SRV setpoint tolerance relaxation.

- ***ECCS/Loss-of-Coolant Accident (LOCA) performance*** – An evaluation concluded that an increase in SRV setpoint tolerances from ±1% to ±3% would not affect the limiting event and would have a negligible impact on the fuel assembly peak cladding temperature (PCT) for non-limiting events.

- ***High pressure emergency systems performance*** – An evaluation has determined that the high pressure injection systems (HPCI, RCIC, & SLC) have the capability of meeting their design basis functional requirements under the increasing reactor pressure conditions imposed by the setpoint tolerance relaxation to ±3%, either directly, with relaxation of the required injection rate, or re-calibration of the turbine controller to increase the maximum rated turbine/pump speed and pump dynamic head.

- ***Containment pressures, temperatures and SRV loads*** - The SRV setpoint tolerance relaxation to ±3% does not affect design basis accident response and will not significantly affect long term containment pressure and temperature response. SRV discharge and associated loads may be affected by an increased SRV opening pressure, but have not been evaluated in this report.

- ***Anticipated transients without scram (ATWS) mitigation capability*** – It was found that vessel pressurization remains within the ASME Emergency Code limit of 1500 psig with 2 SRVs OOS. The long-term suppression pool temperature is not significantly affected by changes to SRV setpoint tolerance.

In summary, there is no adverse impact on the safety-related systems or plant safety due to the relaxation of SRV setpoint tolerance from the current ±1% value to ±3%. The minimum number of operable SRVs specified in the Technical Specifications should be changed from eleven to twelve SRVs (the allowable number of SRVs assumed inoperable is changed from three to two) based on the transient and ATWS overpressure events analysis.

These conclusions are applicable to both Units 1 and 2.

1.0 INTRODUCTION

1.1 PURPOSE

This report documents the results of an engineering evaluation performed to support Technical Specifications changes of the Safety/Relief Valve (SRV) setpoint tolerance from current ±1% to the proposed ±3% for Limerick Generating Station (LGS) Units 1 and 2.

The current performance requirements for the LGS SRVs are discussed in Section 1.3. The present performance requirements pertinent to this analysis are identified, as well as the associated limitations and the remedial actions for exceeding the limitations. Section 1.4 discusses the proposed performance requirement changes, the associated limitations and the analyses required to support the proposed changes. A comparison of the present and proposed performance requirements is shown in Table 1-1.

1.2 BACKGROUND

LGS Units 1 and 2 have virtually identical system configurations and share similar thermal hydraulic and transient behavior characteristics. Therefore, the impact of SRV setpoint tolerance relaxation is expected to be the same for both LGS Units 1 and 2. Consequently, the plant specific analyses performed for Unit 2 is considered representative to quantify impact with SRV setpoint tolerance relaxation for both units.

The pressure relief system for the nuclear reactor pressure vessel at LGS Units 1 and 2 consists of fourteen (14) SRVs located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The SRVs provide the following two primary functions:

- **Overpressure safety/relief function**: The SRVs, functioning in the self-actuated safety mode, open to prevent reactor vessel overpressurization.

- **Depressurization operation**: The Automatic Depressurization System (ADS) function is performed by five of the SRVs, which open automatically by way of a pilot air system. It is considered to be part of the Emergency Core Cooling System (ECCS) and is utilized for events involving small breaks in the reactor vessel process barrier.

1.3 PRESENT PERFORMANCE REQUIREMENTS

1.3.1 SRV Setpoint Tolerances

From Reference 1, the current SRV configuration and the associated nominal opening setpoint values for LGS Units 1 and 2 are as follows:

Safety Relief Valve:	4 Valves @ 1170 psig ± 1%
	5 Valves @ 1180 psig ± 1%
	5 Valves @ 1190 psig ± 1%

A narrow ±1% tolerance band on the SRV nominal setpoints was originally adopted to develop the Technical Specifications due to the acceptance criterion originally defined by the American Society of Mechanical Engineers (ASME), Section III, NB-7000. Section 3/4.4.2 of the current Technical Specifications states that the allowable setpoint variance for each SRV shall be ±1%. The original Technical Specification criterion used was for consistency with the NB-7000 criterion and consistency with the SRV setpoint upper limit used in the original overpressure protection analysis. The ASME has since revised the criterion for demonstrating acceptable in-service SRV operational readiness from ±1% to ±3% [Reference 2].

A valve opening setpoint variance greater than ±1% of the nominal setpoint adds to the number of reportable events and diverts utility work force resources. Although the ±1% tolerance is specified in the Technical Specifications and has been used in plant safety evaluations, it does not represent the limiting setpoint required to ensure plant safety. Several BWRs have experienced SRV setpoint drift more than the Technical Specification limitations. In each case, safety evaluations were performed on a plant-specific and cycle-specific basis, demonstrating that setpoint drift did not compromise plant safety. The need for such safety evaluations may be minimized by increasing the setpoint tolerance for the SRVs as specified in the Technical Specifications for LGS Units 1 and 2.

1.4 PROPOSED PERFORMANCE REQUIREMENT CHANGES

This section discusses the effect of each set of the proposed performance requirement changes and the analyses necessary to support the changes. The present and proposed SRV performance requirement changes are shown in Table 1-1.

The ASME has increased the acceptance criterion for relief valves in-service setpoint tolerance from ±1% to ±3% [Reference 2]. The BWROG has also recommended that the SRV's pressure setpoint tolerance for the Technical Specifications be changed from ±1% to ±3% [References 8, 13]. Consequently, as long as the maximum valve opening pressure remains below the nominal +3% limit and the minimum valve opening pressure remains above the -3% limit for each valve group, the plant will satisfy the code requirement and the valves can be considered capable of performing their pressure relief function.

1.4.1 SRV Installation Tolerance

Normal maintenance includes periodic refurbishment of SRVs. Prior to placing refurbished valves into service, the valve opening setpoints must be adjusted to be within ±1% of their nominal setting. This performance requirement is not changed.

1.4.2 SRVs Out-of-Service

The current LGS Technical Specifications allows up to three (3) SRVs to be declared out-of-service (OOS) [Reference 3] based on the vessel overpressure event analysis with the current ±1% setpoint tolerance. The allowable number of SRVs OOS with the proposed ±3% tolerance is re-evaluated based on the analysis supporting ECCS-LOCA, thermal limits, HPCI performance, RCIC performance, SLCS performance, and ATWS mitigation capabilities vessel overpressure event analysis.

Table 1-1

COMPARISON OF PRESENT TO PROPOSED PERFORMANCE REQUIREMENTS

Performance Requirement	Present Limit	Proposed Limit
Opening pressure at which the SRVs are capable of performing their intended function (i.e., operable).	±1%	±3%
Opening pressure at which licensing basis other than reload analyses have been performed.	±1%	±3%
Opening pressure at which the cycle-specific reload licensing basis analyses have been performed.	±1%	±3%
Tolerance beyond which additional valve testing is required as demonstrated by surveillance testing.	±1%	±3%
Tolerance on the as-left SRV setting prior to the valve being returned to service.	±1%	±1%
Number of SRVs assumed OOS based on the most limiting case.	3	2

2.0 ANALYSIS OVERVIEW

This section identifies the analyses that may be affected by the proposed changes in SRV setpoint tolerance requirement. The cycle-dependent analyses performed in the following sections assume that the plant operating parameters and the core design are consistent with the LGS Unit 2 Cycle 4 licensing calculations.

Based on a review of the UFSAR and Reference 13, the following safety and regulatory aspects are identified as potentially being affected as a result of the SRV setpoint tolerance increase to ±3%:

- Vessel overpressure protection.
- Fuel thermal limits during anticipated operational occurrences.
- ECCS/Loss-of-Coolant Accident (LOCA) performance.
- High pressure emergency systems performance.
- Containment pressures, temperatures, and loads.
- Anticipated transients without scram (ATWS) mitigation capability.

Each applicable safety and regulatory aspect identified in the above listed items except for containment loads was reviewed to determine the acceptability of increasing the SRV opening setpoint tolerance to ±3%. The containment loads is beyond GE's scope. Current containment loads evaluation was performed by PECO and is documented in the BOP Power Rerate Engineering Report PECO-TR-3. The current LGS Technical Specification allows any three SRVs to be OOS with ±1% SRV setpoint tolerance. These analyses were performed to determine whether the same number (3) of SRVs can be OOS with the proposed SRV setpoint tolerance relaxation. The alternate operating modes, including Maximum Extended Operating Domain, Increased Core Flow and Single-Loop Operation, were considered in determining the most restrictive analytical condition (i.e., most limiting operating mode) for performing the analysis associated with this proposed technical specification change. For each analysis, the most limiting operating mode permitted by LGS Units 1 and 2 operating license was assumed. Discussion of the individual analyses and evaluations is presented in the following sections of this report and the conclusions are summarized in Table 2-1.

Table 2-1
ANALYSES PRESENTED IN THIS REPORT AND ACCEPTANCE CRITERIA

Item	Section	Acceptance Criteria
Vessel Overpressurization	3.0	Peak vessel bottom pressure less than ASME Code limit of 1375 psig and steam dome pressure less than Technical Specification Limit of 1325 psig
Thermal Limits	4.0	MCPR safety limit protection maintained
ECCS/LOCA Performance	5.0	PCT < 2200°F
High Pressure System Performance Evaluation	6.0	Adequate design basis injection at +3% above lowest SRV setpoint pressure
Containment Evaluations	7.0	Containment pressure and temperature within acceptance limits
ATWS Mitigation	8.0	Peak vessel bottom pressure less than ASME Service Level C Limit of 1500 psig, and suppression pool temperature less than specified limits

3.0 VESSEL OVERPRESSURE PROTECTION ANALYSIS

One of the design requirements of the SRVs is to limit the pressure rise to less than 110% of the design pressure (1375 psig) for events with a frequency of occurrence in the ASME Upset (Service Level B) category (i.e., anticipated operational occurrences or "transients"). The limiting overpressure event for LGS Units 1 and 2 is the Main Steamline Isolation Valve (MSIV) Closure with Flux Scram event. This event assumes the failure of the MSIV position switches to initiate a scram. The reactor is shut down by the second scram signal, the high neutron flux scram caused by the vessel pressurization and the resultant collapse of moderator voids within the reactor core.

The current Technical Specifications basis of the vessel overpressure protection analyses for LGS Units 1 and 2 takes credit for 11 out of 14 SRVs and with the SRV setpoints at 1% above nominal. To justify the increase in setpoint tolerance, the analyses are performed with the assumption that all the SRV opening setpoints have drifted above their nominal trip setpoint by 3%.

3.1 OVERPRESSURE PROTECTION ANALYSIS ASSUMPTIONS

The GE thermal-hydraulic and nuclear kinetics coupled transient code, ODYN, was used to calculate the system response and peak vessel pressure. ODYN is a one-dimensional core model that combines neutron kinetics, thermal-hydraulic, and plant-specific characteristics to evaluate rapid pressurization events. ODYN has been approved by the NRC for this application.

3.2 OVERPRESSURE PROTECTION ANALYSIS RESULTS

3.3 CONCLUSIONS

The analysis shows that LGS Units 1 and 2 limiting overpressure event of MSIV Closure with Flux Scram, along with 2 SRVs OOS and with relaxation of SRV setpoint tolerance from ±1% to ±3%, would not violate the ASME required peak vessel bottom pressure limit of 1375 psig and vessel steam dome pressure is within the Technical Specification Steam Dome Pressure Limit of 1325 psig.

4.0 THERMAL LIMITS

The effect of transients on fuel thermal limits is one of the important considerations for relaxation of the SRV setpoint tolerance. The minimum critical power ratio (MCPR) is the most significant thermal limit for this evaluation. The MCPR for the event must be maintained above the MCPR safety limit. A review of the LGS Units 1 and 2 analysis identified the transients that have the greatest potential effect on fuel thermal limits.

4.1 ANALYSIS AND RESULTS

4.2 CONCLUSIONS

The analysis shows that the thermal limits of the limiting anticipated operational occurrence (or transient event), the Feedwater Controller Failure or the Load Rejection with Failure of the Bypass System (LRNBP) events, would not be affected by the relaxation of SRV setpoint tolerance from ±1% to ±3%. Further, other transient events remain non-limiting and bounded by the above events.

5.0 ECCS/LOCA PERFORMANCE EVALUATION

The LGS Units 1 and 2 LOCA analysis [Reference 5] has been reviewed to determine the effect of increased variation in the SRV opening pressures and the effect of the number of SRVs inoperable on the Emergency Core Cooling System (ECCS) performance.

The ECCS are designed to provide adequate core cooling during a postulated LOCA by limiting the calculated PCT to below the requirements of 10CFR50.46 (i.e., to less than 2200°F). A change in the SRV opening pressures can only affect response of the pipe break events for which SRV actuation occurs. An assessment of these events was performed to determine the effect of valve actuation at ±3% around the setpoint. The intent of this assessment was to demonstrate that the limiting break (i.e., the one yielding the highest PCT) remains unaffected by these changes and that the effect on the other "breaks", which have a lower PCT, would not cause them to become the limiting break.

The following postulated pipe break scenarios were considered:

- Limiting break LOCA (licensing basis event)
- Small break LOCA
- Steamline break outside the containment.

5.1 LIMITING BREAK LOCA

Based on a review of the results of the Reference 5 analyses, the limiting break for LGS Units 1 and 2 is the design basis accident (DBA) recirculation line break. For this type of event, the reactor vessel depressurizes very rapidly through the break itself. Because of the rapid vessel depressurization, SRV actuation will not occur. Therefore, an change in the SRV opening setpoint or number of SRVs OOS does not have any adverse impact on the limiting break analysis results.

5.2 SMALL BREAK LOCA

For a postulated small break LOCA, an increased SRV opening pressure due to +3% setpoint drift will not have a significant effect on the overall system response. The small break events are not limiting in comparison to the large break. Based on the above discussion, the small break LOCA will remain non-limiting.

5.3 STEAMLINE BREAK OUTSIDE THE CONTAINMENT

In this event, a double-ended guillotine break of one main steamline occurs outside the containment. The steamline break LOCA event will remain non-limiting with the proposed SRV setpoint tolerance relaxation.

5.4 CONCLUSIONS

The relaxation of the SRV setpoint tolerance has no adverse impact on DBA LOCA analysis since the SRVs do not actuate for this event. The response of other breaks in which SRVs actuate are insensitive to variation in SRV opening pressure and the number of SRVs OOS. Further, other non-limiting breaks which have lower PCTs will still remain bounded by DBA LOCA case.

6.0 HIGH PRESSURE SYSTEM PERFORMANCE

This section documents the results of evaluations of the impact of the SRV opening setpoint tolerance changes on the performance of the following high pressure systems:

- High Pressure Coolant Injection (HPCI)
- Reactor Core Isolation Cooling (RCIC)
- Standby Liquid Control System (SLCS)

This evaluation is based on the current design basis requirements for these systems. Increasing SRV opening pressure is limiting for system performance evaluations. Changing the upper SRV setpoint tolerance from +1% to +3% above the nominal setpoint will increase the maximum reactor pressure in which these high pressure systems must operate which could potentially cause a reduction in the performance capability of these systems.

6.1 HIGH PRESSURE SYSTEM EVALUATION

6.1.1 HPCI SYSTEM

The High Pressure Coolant Injection (HPCI) System, an ECCS, is designed to provide sufficient core cooling and prevent excessive fuel cladding temperature in the event of a small break LOCA that does not inherently depressurize the reactor rapidly enough to permit timely operation of the low pressure ECCS. The HPCI System accomplishes this function by injecting coolant makeup water into the reactor pressure vessel using a turbine driven pump. During a LOCA event, the HPCI System is automatically initiated upon receipt of either a low reactor water level or a high drywell pressure signal. The HPCI System also provides coolant flow for other events where Feedwater (FW) flow is lost (pipe breaks outside containment and transient loss of FW). The Loss of Feedwater events are discussed in Section 6.2.

The current performance design basis for the HPCI System is that it must be capable of injecting design rated flow into the reactor vessel at a maximum reactor pressure equal to the lowest SRV nominal setpoint plus the allowable setpoint tolerance. The HPCI System is also designed to serve as a backup to the RCIC System in the event of a loss of feedwater transient and the vessel becomes isolated from the main condenser. During a reactor isolation with the failure of the RCIC System, the HPCI System is required to maintain the reactor water level above Level 1 while the reactor is maintained at a hot standby condition. The HPCI System injection flow during such an event must be equal to or greater than that of the RCIC System flow rate of 600 gpm. The evaluation showed that the HPCI system has sufficient capacity at the higher reactor pressure to serve as an effective backup to the RCIC system.

Re-calibration of the turbine control system and high steam flow setpoint instrumentation is not required.

6.1.2 RCIC SYSTEM

The Reactor Core Isolation Cooling (RCIC) System is designed to maintain the reactor vessel water level above Level 1 in a transient event that results in the loss of all feedwater flow and the vessel becomes isolated from the main condenser. The system design function is to allow for reactor shutdown by maintaining sufficient water inventory while the reactor pressure is reduced to a level where the shutdown cooling mode of the Residual Heat Removal (RHR) System can be placed into operation.

The RCIC System accomplishes this function by injecting coolant makeup water into the pressure vessel with a turbine driven pump. The RCIC System is designed to automatically start upon receipt of a low reactor water level signal. It is designed to be capable of injecting rated flow of 600 gpm within 30 seconds after onset of the initiating conditions (Tech. Spec. time limit is 55 seconds).

The current performance design basis for the RCIC system is that it must be capable of injecting the system design rated flow into the reactor vessel at a maximum reactor pressure equal to the lowest SRV nominal setpoint plus the allowable setpoint tolerance. The RCIC System performance capability was evaluated for operation at the maximum reactor pressure of 1205 psig (lowest SRV setpoint, 1170 psig, plus 3% setpoint tolerance). It was concluded that with an increase in the maximum rated pump and turbine speed to 4625 rpm, the current RCIC System design is capable of providing its required function under this reactor pressure condition. An increase in the pump rated speed is required in order for the pump to develop the required dynamic head for design basis injection flow rates at the higher pressures. Re-calibration of the turbine control system is required. System operation at the increased reactor pressure will result in an increase in the maximum required steam flow to the turbine.

6.1.3 STANDBY LIQUID CONTROL SYSTEM

The Standby Liquid Control System (SLCS) is designed to shut down the reactor from rated power condition to cold shutdown in a postulated event in which all or some of the control rods cannot be inserted or during a postulated ATWS event. The SLCS accomplishes this function by pumping a sodium pentaborate solution into the vessel within a prescribed boron injection rate in order to provide neutron absorption and achieve a subcritical reactor condition.

The current performance design basis for the SLCS is that it must be capable of injecting the system design rated flow into the reactor vessel using a single SLC pump at a maximum reactor pressure equal to the lowest SRV nominal setpoint plus the allowable setpoint tolerance, in the event in which the operator cannot manually insert a sufficient number of control rods to bring the reactor to subcritical conditions. The SLCS performance capability was evaluated for pump operation at the maximum reactor pressure of 1205 psig for the 3% setpoint tolerance relaxation. The maximum operating pressure for the system is limited by the setpoint for system relief valves located at the discharge of each pump (Unit 1 = 1400 psi, Unit 2 = 1375 psi). To prevent system relief valve leakage or lifting during operation, a relief valve pressure margin of 57 psi (from nominal setting) is necessary for the LGS SLCS. With a single pump in operation (43 gpm), the calculated maximum pump discharge pressure of 1281 psig results in a system relief valve margin of 119 psi for Unit 1, and 94 psi for Unit 2 . These pressure margins exceed the minimum LGS value of 57 psi needed to assure full SLCS injection into the reactor.

Injection of the solution using two pumps is required to control reactivity in the event of a postulated ATWS event where the control rods cannot be inserted to maintain subcritical conditions. ATWS analyses provide a best guess estimate of the expected response of the reactor based on a nominal SLCS flow rate of 86 gpm, assuming nominal SRV setpoint pressures. Because the proposed change to the Safety Relief Valve setpoint tolerance does not result in a change to the nominal setpoint, a specific reevaluation of the SLCS performance capability in relationship to the ATWS events is not required and the SLC system design requirements will not be changed as the result of the increase in setpoint tolerance. However, a qualitative evaluation was performed to determine the general impact on the SLCS.

From this evaluation it was concluded that the current SLCS design is capable of meeting both the system design basis and ATWS requirements for injection with the SRV setpoint tolerance program.

6.2 LOSS OF FEEDWATER EVENT

During a Loss of Feedwater event, the reactor water level decreases steadily. When the reactor water level decreases to Level 2, the reactor recirculation pumps are tripped and HPCI and RCIC pumps are signaled to start. The RCIC System or HPCI System (backup to the RCIC System) is capable of maintaining the reactor water level. The analysis for the loss of feedwater event indicates that the reactor water level will not be reduced to Level 1 (the MSIV isolation setpoint) and therefore will not result in vessel isolation. Consequently, during this event, the reactor does not pressurize, and thus requires no SRVs actuation. Hence, SRV setpoint tolerance relaxation has no effect on a Loss of Feedwater event where reactor isolation does not occur.

In the event of a reactor isolation that results in loss of feedwater, the RCIC system and/or the HPCI system, operate(s) to maintain the reactor water level. As described in Section 6.1, the HPCI and RCIC systems performance capability can be restored to its design basis performance so that its response during a reactor isolation is not affected by the SRV setpoint tolerance relaxation. Therefore, with appropriate modifications identified in Section 6.1, SRV setpoint tolerance to ±3% does not impact the performance of HPCI/RCIC during a loss of Feedwater event with reactor isolation.

6.3 CONCLUSIONS

The performance capability of the high pressure injection systems (RCIC, HPCI, and SLCS) have been evaluated for a reactor pressure condition, up to 1205 psig, based on the lowest SRV setpoint plus 3% tolerance. It was concluded that the high pressure systems are capable of performing their intended design basis and transient functions under the SRV setpoint tolerance relaxation program, through relaxation of the required injection rate for the HPCI System, and by re-calibration of the turbine controller for an increased maximum rated pump/turbine speed and pump dynamic head for the RCIC System. No changes were required by the SLCS.

7.0 CONTAINMENT EVALUATION

The increase in the SRV setpoint tolerance to ±3% was assessed to determine the potential impact on the containment design limits. The two primary areas of concern for the containment structures are (1) the pressure and temperature response and (2) the containment hydrodynamic loads. This discussion is limited to an evaluation of the pressure and temperature response. The hydrodynamic load calculation input, i.e., SRV flow rate, is provided for use by others in performing the evaluation.

SRV actuation exerts pressure and thrust loads on the SRV discharge piping and T-quencher. In addition, the expulsion of water and then air/steam into the suppression pool through the T-quencher results in pressure and drag loads on submerged structures. These SRV discharge loads are potentially affected by an increase in the SRV flow rate due to an increase in the SRV setpoint tolerance. To assist in the determination of hydrodynamic loads, the maximum reactor dome pressure and SRV discharge flow rates for combinations of the SRV setpoint tolerance and number of SRVs out-of-service were calculated.

7.1 CONTAINMENT PRESSURE AND TEMPERATURE

The most limiting event in terms of peak containment pressure and temperature response as well as peak suppression pool temperature is the design basis accident (DBA) LOCA, a double-ended guillotine break of the recirculation line. Relaxation of the SRV setpoint tolerance has no effect on this event because the vessel depressurizes without any SRV actuation. Therefore, there is no impact on the DBA-LOCA containment peak pressure and temperature or on the peak DBA-LOCA suppression pool temperature.

Small steamline breaks can result in high drywell temperature conditions which can last for relatively long time periods because the vessel remains at high pressure for a longer period than for the DBA [Reference 5]. For small steamline breaks with SRV actuation, the peak drywell temperature occurs relatively late in the event following many SRV actuation cycles. The containment temperature is primarily dependent on the total inventory loss. The inventory loss relies on the decay heat which remains unaffected by the SRV setpoint tolerance relaxation. Therefore, an increase in the SRV setpoint tolerance to ±3% will not have an impact on the peak drywell temperature for small steamline breaks. Similarly, the number of SRVs OOS does not affect the overall inventory loss and, hence, has a negligible impact.

7.2 SRV Flow Rates Analysis Assumptions

When SRVs actuate, the ambient reactor pressure forces first water and then air/steam into the suppression pool. The highest SRV discharge loads occur during the first SRV actuation when water initially in the SRV piping is expelled into the suppression pool. Once the SRVs clear, steam from the reactor is forced into the suppression pool. The highest discharge pressure is equal to the highest SRV nominal setpoint plus any setpoint tolerance.

7.3 SRV Flow Rates Analysis Results

7.4 CONCLUSIONS

The increase of the SRV opening setpoint tolerance from ±1% to ±3% does not affect the DBA LOCA response of the containment in terms of containment pressure and temperature. The increase in the SRV opening setpoint tolerance from ±1% to ±3% has negligible effect on the total inventory loss; thus, the overall effect on the containment pressure and temperature is negligible. The SRV discharge loads may be affected by the SRV setpoint tolerance change. SRV flow rates have been provided so that this may be evaluated by others.

8.0 ATWS MITIGATION ANALYSIS

8.1 OVERVIEW

LGS Units 1 and 2 includes safety features to mitigate the consequences of an Anticipated Transient Without Scram (ATWS). Although evaluation of plant response for ATWS events is beyond design basis, and these events are assessed on the best-estimate basis, . the impact of increasing Safety Relief Valve setpoint tolerance relaxation to +3% (the maximum tolerance limit is most limiting for vessel overpressure event) is evaluated for peak vessel overpressure and peak suppression pool temperature for ATWS events.

The evaluations have been performed by using the ODYN computer code. The use of ODYN for the short-term pressurization response of this event is more appropriate than is the use of REDY. ODYN is a one dimensional model, whereas REDY uses point kinetics modeling. Thus, axial power variation is utilized by ODYN, but not by REDY. Additionally, ODYN simulates pressure waves and therefore improves the accuracy of pressure rate and water level response of the model. NRC has approved the use of ODYN for ATWS evaluations [Reference 14].

8.2 ATWS ANALYSIS METHOD

8.3 INPUTS AND ASSUMPTIONS

8.4 ATWS ANALYSIS RESULTS

8.5 CONCLUSIONS

Based on the analysis, it is concluded that changing SRV setpoint tolerance to ±3% of the nominal setpoint would not adversely impact the vessel the suppression pool temperature criteria for the limiting ATWS event. With up to 2 SRVs OOS, the vessel overpressure criteria of 1500 psig can be satisfied.

9.0 CONCLUSIONS

Based on the evaluations and analyses performed and described in the foregoing sections of this report, it has been determined that all the proposed performance objectives listed in Table 1-1 have been satisfied for LGS Unit 1 and 2 with the exception of containment loads associated with SRV discharge, which have not been evaluated herein. The evaluations support the basis for SRV setpoint tolerance relaxation from ±1% to ±3% of the nominal setpoint.

The ECCS/LOCA performance and ATWS mitigation analysis are cycle independent governed by GESTAR Amendment 22. The high pressure system performance and containment response analysis are cycle independent. The vessel overpressurization analysis and fuel thermal limits are evaluated at each cycle for SRV setpoint tolerance at +3%. The effect of -3% drift has been generically evaluated in this document and therefore is cycle independent.

These conclusions are applicable to both LGS Units 1 and 2.

10. REFERENCES

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2. ANSI/ASME OM-1-1981, as referenced in Subsection IWV-3500 of the ASME Code, Section XI, 1986 Edition.

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