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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE NO. NPF-39

PECO ENERGY COMPANY

LIMERICK GENERATING STATION, UNIT 1

DOCKET NO. 50-352

1.0 INTRODUCTION

By letter dated January 12, 1999, as supplemented January 29, March 10, and September 20, 1999, PECO Energy Company (the licensee) submitted proposed changes to the Limerick Generating Station (LGS), Unit 1, Technical Specifications (TSs). The requested changes would revise TS Section 3/4.4.2, "Safety/Relief Valves," and TS Bases Sections B 3/4.4.2, B 3/4.5.1, and B 3/4.5.2 to increase the allowable as-found main steam safety relief valve (SRV) code safety function lift setpoint tolerance from plus or minus 1percent to plus or minus 3 percent. Also, the required number of operable SRVs in operational conditions 1, 2 and 3 will be increased from 11 to 12. In the September 20, 1999, letter, the licensee committed to ensure that the 3 percent lift setpoint tolerance value for the SRVs will be incorporated into the LGS Unit 1 core reload analysis for Cycle 9 operation. The January 29, March 10, and September 20, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original Federal Register notice.

2.0 BACKGROUND

It is stated in 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design," that "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

The proposed change does not alter the SRV safety lift setpoints, relief setpoints, or the SRV lift setpoint test frequency. Also, the proposed change requires the as-left safety valve function settings to be within plus or minus 1% of the specified nominal lift setpoints prior to installation. The NRC staff has previously granted approval to individual boiling-water reactors (BWRs) to increase the as-found SRV tolerance to 3 percent. The basis for the approval was the staff's safety evaluation report (SER), dated March 8, 1993, for a licensing topical report (NEDC-31753P) evaluating the setpoint tolerance increase. The staff's SER included six conditions which must be addressed on a plant-specific basis for licensees applying for the increased SRV setpoint tolerance:

(a) Transient analysis of all abnormal operational occurrences as described in NEDC-31753P "BWROG In-Service Pressure Relief Technical Licensing Topical Report," should be

performed utilizing a plus or minus 3 percent tolerance for the safety mode of spring safety valves (SSVs) and SRVs. In addition, the standard reload methodology (or other method approved by the staff) should be used for this analysis.

(b) Analysis of the design basis over pressurization event using the 3 percent tolerance limit is required to confirm that the vessel pressure does not exceed the American Society of Mechanical Engineers (ASME) pressure vessel code upset limit.

(c) The plant-specific analysis described in items (a) and (b) should assure that the number of SSVs and SRVs, and relief valves (RVs) included in the analyses correspond to the number of valves required to be operable in the TSs.

(d) Reevaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc,), motor-operated valves, and vessel instrumentation and associated piping must be completed, considering the 3 percent tolerance limit.

(e) Evaluation of the plus or minus 3 percent tolerance on any plant-specific operating modes (e.g., increased core flow, extended operating domain, etc.) should be completed.

(f) Evaluation of the effect of the 3 percent tolerance limit on the containment response during loss- of-coolant accidents and the hydrodynamic loads on the SRV discharge lines and containment should be completed.

3.0 EVALUATION

The safety objective of the SRVs is to prevent overpressurization of the nuclear system. This protects the nuclear system process barrier from failure which could result in the uncontrolled release of fission products. The pressure relief system at Limerick Unit 1 includes fourteen SRVs, arranged into three setpoint groupings: one group of SRVs (4) set at 1170 psig, a second group of SRVs (5) set at 1180 psig, and a third group of SRVs (5) set at 1190 psig. The existing TS provides a plus or minus 1 percent as-found tolerance and plus or minus 1 percent as-left setpoint tolerance. The proposed modifications would provide a plus or minus 3 percent as-found tolerance and plus or minus 1 percent as-left setpoint tolerance. The licensee's submittal was evaluated against the generic SER described above.

3.1 Transient Analysis/Reload Methodology

The licensee must consider the impact of the tolerance increase on abnormal operational transients (AOTs). Limerick Unit 2 analysis (cycle 4 reload analysis) of AOTs has been conducted utilizing the 3 percent tolerance and with all 14 SRVs in service. The transients which generate the limiting decrease in a critical power ratio are the load rejection without turbine bypass event and feedwater controller failure of the bypass system. The analysis showed that the thermal limits of the limiting transient would not be affected by the relaxation of SRV setpoint tolerance. Further, other transient events remain non-limiting and bounded by the above event. The NRC-approved licensing analysis methodology, "Core Operating Limit Report for Limerick Generating Station, Unit 2, Reload 4 Cycle 5, Rev. 2," dated November 1998, was used for the analysis. The above mentioned results are based on an anaylsis for LGS, Unit 2,

Cycle 4. However, the conclusions are generic for both LGS Unit 1 and 2 and can be applied to both due to the virtually identical system configuration and shared similar thermal hydraulic and transient behavior characteristics.

3.2 Analysis of the Design Basis Overpressurization Event

The licensee is required to reevaluate the limiting design basis pressurization transient using the 3 percent tolerance limit to confirm that the vessel pressure does not exceed the ASME pressure vessel code upset limit. The ASME Boiler and Pressure Vessel Code Section III permits pressure transients up to 10 percent over design pressure (110 percent x 1250 psig = 1375 psig). The limiting pressurization AOT analyzed is a main steam Isolation valve (MSIV) closure event occurring at the end of full power life without credit for a reactor trip on MSIV position sensing. The licensee analyzed the MSIV closure event using the staff-approved model ODYN with the 3 percent tolerance and calculated the maximum vessel pressure to be 1318 psig assuming two SRVs are inoperable. This is within the 1375 psig ASME limit, and is acceptable to the staff.

3.3 TS Operability Statement for SRVs

The licensee has stated that plant-specific overpressure analyses have been conducted with the number of SRVs included in the analyses corresponding to the number of valves required to be operable in the TSs. The analysis took credit only for 12 of the 14 SRVs required by the TSs. This is acceptable to the staff.

The LGS Unit 1 SRVs are currently Target Rock 2-stage valves which have experienced several occurrences of positive setpoint drift in excess of the plus or minus 3 percent used in the licensee's analysis to support this TS change. The cause of the drift has been determined to be corrosion bonding between the pilot disk and its seat. As a corrective action, the licensee (in Licensee Event Report (LER) 98-008-01) has stated that it has installed platinum ion-beam implanted pilot disks in 7 of the 14 plant SRVs. The platinum ion-beam implanted pilot valve disks have significantly reduced the SRV setpoint drift experienced at other BWR plant sites having SRVs of the same pilot stage design as those at LGS Units 1 and 2. Further, the licensee stated in LER 98-008-01, that the 2-stage pilots are to be removed and 3-stage pilots are to be installed in all 14 plant SRVs for both LGS Units 1 and 2. This modification was completed during the April 1999 refueling outage for LGS Unit 2 and is also scheduled for LGS Unit 1 spring 2000 refueling outage. The Target Rock 3-stage SRVs installed at other BWR sites have experienced significantly less setpoint drift than 2-stage SRVs. Therefore, the staff finds that the licensee's corrective actions are consistent with the proposed TS change to the SRV setpoint tolerance.

3.4 Reevaluation of the performance of High Pressure Systems

The licensee must also reevaluate the performance of high pressure systems (pump capacity, discharge pressure, etc.), considering the 3 percent tolerance limit. LGS Unit 1 has three systems which are required to inject liquids into the vessel at high pressure conditions: High Pressure Coolant Injection System (HPCI), Reactor Core Isolation Cooling (RCIC) and Standby Liquid Control System (SLCS). The most significant impact is the increased reactor pressure

specified for systems operation. The systems' performances were evaluated for increasing the reactor pressure to 1205 psig from 1182 psig. The licensee concluded that the HPCI pump discharge piping would exceed currently specified design values with the system design flow of 5400 gpm at reactor pressure of 1205 psig if the pump discharge valve is inadvertently closed. Therefore, the HPCI system design basis has been changed to inject a flow of only 5000 gpm to the vessel at the pressure between 1182 and 1205 psig. There are now two design flow requirements: 5600 gpm at reactor pressures between 200 psig and 1182 psig and 5000 gpm at reactor pressures between 1182 psig and 1205 psig.

The RCIC turbine/pump maximum speed is increased from 4575 rpm to 4625 rpm in order for the RCIC system to perform at the new maximum reactor operating pressure. The increased speed reduces the over speed margin from 123 percent to 122.1 percent. This reduction in margin is acceptable due to the system modifications to the turbine start feature. The SLCS system was determined to have the capability to inject boron into the vessel at its design flow rate at the higher reactor pressures.

3.5 Evaluation of Motor-Operated Valves and Piping

The licensee stated that the impact on motor-operated valves (MOVs) due to the potential for increased reactor pressure as a result of the increase in SRV setpoint tolerance was evaluated and was determined to be acceptable. The licensee stated that the plant Generic Letter (GL) 89-10 Program currently uses SRV nominal setpoints for differential pressure determinations for valves in which reactor pressure at the SRV setpoint is limiting. The licensee also stated that the operating pressure of the RCIC system was changed by the increased turbine/pump rated speed and that the impact of this change on the MOVs has been evaluated using the guidance in the GL 89-10 Program and determined to be acceptable. The staff finds that meeting the requirements of the GL 89-10 Program ensures the design-basis capability of the MOVs.

The licensee also evaluated the effects of the high pressures associated with the increased setpoint tolerance on the instrumentation and piping for the systems. The licensee determined that no changes to instrumentation will be required. The staff finds that this is acceptable.

3.6 Simmer Margin

The potential concern regarding simmer margin, the difference between the maximum normal operating pressure and the lowest SRV setpoint, is that with less simmer margin, there is less seating force and there may be an increased tendency for the valves to leak or inadvertently open. Using the proposed plus or minus 3 percent setpoint tolerance, the simmer margin would be a minimum of 89.9 psi. The minimum simmer margin using the current plus or minus 1 percent setpoint tolerance is 113.3 psi. The licensee stated that this reduction in the minimum simmer margin has been evaluated by the SRV manufacturer for both the 2-stage and 3-stage Target Rock SRVs and determined to be acceptable. The staff finds that meeting the SRV manufacturer's recommendations is acceptable regarding simmer margin, and that the licensee has adequately addressed this concern.

3.7 Alternate Operating Modes

The licensee must also evaluate the increased tolerance on any plant-specific alternate operating modes (e.g., increased core flow, maximum extended load line limit, single loop operation, etc.) The analyses included evaluations for the currently approved operating domains: Maximum Extended Load Line Limit, and Increased Core Flow and Single Loop Operation. This is acceptable to the staff.

3.8 Containment Response/Hydrodynamic Loads

The licensee must also evaluate the effect of the increased tolerance limit on (1) the containment hydrodynamic loads during loss-of-coolant accidents (LOCAs) and (2) the hydrodynamic loads on the SRV discharge lines and the suppression chamber.

The licensee examined the potential effects of the proposed amendment on the containment design limits. The containment design basis accident is a double-ended break at the suction of a recirculation pump. For this event, the reactor coolant system (RCS) depressurizes very rapidly and thus, the SRVs are not challenged. Also, the RCS inventory and primary system heat sources that would contribute to the containment mass and energy are not increased. The setpoint tolerance thus has no effect on the capability of the containment to perform its design basis safety function (i.e., the containment peak temperature and pressure loads would not be adversely affected). The staff notes that small-break LOCAs also would not lead to increased RCS pressure and subsequent SRV challenges.

An increase in SRV setpoint tolerance involves a potential increase in SRV discharge hydrodynamic loads on the SRV discharge piping and the containment structures. The licensee stated that the calculated SRV discharge loads include an additional 5 percent conservatism, which is due to a factor of 1.05 included in the current calculated SRV flow.

The plant-specific quencher loads are based on actual test data based on a vessel pressure of 1276 psig. Since the highest nominal SRV setpoint of 1190 psig yields a maximum reactor vessel pressure of 1226 psig, the licensee's proposed increase in setpoint will not affect the basis of the plant's hydrodynamic loads design.

3.9 ECCS-LOCA

GE reviewed the LOCA analysis in the LGS Unit 1 Updated Safety Analysis Report to determine the effect of an increase in SRV opening pressures on emergency core cooling system performance. The limiting break LOCA, the design basis accident recirculation break, the small-break LOCA and the steamline break outside containment events were evaluated to determine the effects of the increased SRV setpoint tolerance. GE performed an analysis with the HPCI reduced flow of 5000 gpm at reactor pressure of 1205 psig. Both the current HPCI system design flow of 5600 gpm at reactor pressures between 200 to 1182 psig, and the proposed HPCI design flow of 5000 gpm at reactor pressures between 1182 psig and 1205 psig satisfy the requirements of 10 CFR 50.46. The acceptance criteria given in 10 CFR 50.46 are still satisfied for all break sizes and locations and hence the setpoint tolerance change for LOCA considerations is acceptable.

3.10 Anticipated Transient Without Scram (ATWS)

The limiting event for LGS Unit 1 for ATWS conditions is the pressure regulator failure open transient without scram (PREGO). MSIV closure and PREGO are similar events and either event has the potential to result in the maximum calculated reactor pressure. For this reason both events were considered. The MSIV closure event assumes that all MSIVs close simultaneously while the unit is operating at rated conditions. The PREGO event assumes that the pressure regulator fails open, resulting in maximum steam demand. This maximum demand results in a reduced pressure causing the MSIVs to close on low steamline pressure. Therefore, the MSIV closure during a PREGO event occurs at conditions other than rated steam conditions. Using the staff-approved ODYN code and assuming two SRVs inoperable, the analysis shows that the vessel pressure reaches a maximum of 1468 psig, which is within the vessel overpressure criterion of 1500 psig for ATWS events. The long-term effect on suppression pool temperature due to 3 percent SRV tolerance is negligible because there is little change in the total energy discharged to the pool. This is acceptable.

3.11 TS Changes

TS 3.4.2 - The number of SRVs required to be operable is increased from 11 to 12 since credit is taken for 12 valves in the analysis. This is acceptable. The setpoint tolerance in TS 3/4.4.2 is changed from plus or minus 1 percent to plus or minus 3 percent. This is acceptable as described above. In Surveillance Section 4.4.2.1, the following statement has been added: "All safety valves will be recertification tested to meet a plus or minus 1 percent tolerance prior to returning the valves to service." This is also acceptable.

TS Bases 3/4.4.2 - The number of SRVs required to be operable is increased from 11 to 12 since credit is taken for 12 valves in the analysis.

TS Bases 3/4.5.1 and 3/4.5.2 Emergency Core Cooling System-OPERATING and SHUTDOWN - Last paragraph

The following note is added to the first sentence in the last paragraph of Page B 3/4 5-1 of the TS: "and is capable of delivering at least 5000 gpm between 1182 and 1205 psig." The reduced HPCI flow analysis performed by General Electric verified that LGS Unit 1 is in compliance with 10 CFR 50.46, hence it is acceptable.

3.12 Summary Conclusion

Based on the NRC staff's evaluation of the licensee's submittal, the staff finds the proposed TS changes and associated revisions to the TS Bases acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (64 FR 9194). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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