May 17, 1999

Mr. Garrett D. Edwards
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

SUBJECT: LIMERICK GENERATING STATION (LGS), UNIT 2 - ISSUANCE OF AMENDMENT RE: SAFETY RELIEF VALVE CODE SAFETY FUNCTION LIFT SETPOINT TOLERANCE (TAC NO. MA4562)

Dear Mr. Edwards:

The Commission has issued the enclosed Amendment No. 98 to Facility Operating License No. NPF-85 for the Limerick Generating Station, Unit 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 12, 1999, as supplemented January 29 and March 10, 1999.

This amendment revises 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 1% to plus or minus 3%. Also, the required number of operable SRVs in operational conditions 1, 2, and 3 will be increased from 11 to 12.

A copy of our safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly <u>Federal Register</u> notice.

This completes our efforts on this issue for LGS, Unit 2, and we are, therefore, closing out TAC No. MA4562. As requested, we are deferring our evaluation of LGS, Unit 1, until you have completed the associated core reload analysis for the next Unit 1 refueling outage, which is currently scheduled to commence April 2000.

Sincerely,

Original signed by:
Bartholomew C. Buckley, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-353
Enclosures: 1. Amendment No. 98 to License No. NPF-85
2. Safety Evaluation
cc w/encls: See next page

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Docket File	MO'Brien	WBeckner	JWermiel
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DOCUMENT NAME: AMD4562.WPD *See previous concurrence
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NAME	BBuckley:rb		MO'Brien		JWermiel		GImbro		JHannon				JClifford for	
DATE	5/17/99		5/17/99		4/28/99		4/27/99		4/30/99		5/5/99		5/17/99	

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

May 17, 1999

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Nuclear Group Headquarters
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Docket No. 50-353

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2. Safety Evaluation

cc w/encls: See next page

Limerick Generating Station, Units 1 & 2

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

PECO ENERGY COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 98
License No. NPF-85

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by PECO Energy Company (the licensee) dated January 12, 1999, as supplemented January 29 and March 10, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-85 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 98 , are hereby incorporated into this license. PECO Energy Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to startup following completion of the April 1999 refueling outage for Limerick Generating Station, Unit 2.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Nakoski for

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 17, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 98

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages as indicated. The revised pages are identified by Amendment number and contain marginal lines indicating the area of change.

Remove	Insert
3/4 4-7	3/4 4-7
B 3/4 4-2	B 3/4 4-2
B 3/4 5-1	B 3/4 5-1

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 12 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*#

4 safety/relief valves @ 1170 psig ±3%
5 safety/relief valves @ 1180 psig ±3%
5 safety/relief valves @ 1190 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open safety/relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable acoustic monitors to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 0.20 of the full open noise level## by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 92 days, and a
b. CHANNEL CALIBRATION at least once per 24 months**.

4.4.2.2 At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 24 months, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 54 months. All safety valves will be recertification tested to meet a ±1% tolerance prior to returning the valves to service.

---

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

** The provisions of Specification 4.0.4 are not applicable provided the Surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

\# Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.

\## Initial setting shall be in accordance with the manufacturer's recommendation. Adjustment to the valve full open noise level shall be accomplished during the startup test program.

RECIRCULATION SYSTEM (Continued)

Plant specific calculations can be performed to determine an applicable region for monitoring neutron flux noise levels. In this case the degree of conservatism can be reduced since plant to plant variability would be eliminated. In this case, adequate margin will be assured by monitoring the region which has a decay ratio greater than or equal to 0.8.

Neutron flux noise limits are also established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1-12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels which significantly bound these values are considered in the thermal/mechanical design of GE BWR fuel and are found to be of negligible consequence. In addition, stability tests at operating BWRs have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles of 5-10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Typically, neutron flux noise levels show a gradual increase in absolute magnitude as core flow is increased (constant control rod pattern) with two reactor recirculation loops in operation. Therefore, the baseline neutron flux noise level obtained at a specific core flow can be applied over a range of core flows. To maintain a reasonable variation between the low flow and high flow end of the flow range, the range over which a specific baseline is applied should not exceed 20% of rated core flow with two recirculation loops in operation. Data from tests and operating plants indicate that a range of 20% of rated core flow will result in approximately a 50% increase in neutron flux noise level during operation with two recirculation loops. Baseline data should be taken near the maximum rod line at which the majority of operation will occur. However, baseline data taken at lower rod lines (i.e. lower power) will result in a conservative value since the neutron flux noise level is proportional to the power level at a given core flow.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operates to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 12 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown. The safety/relief valves will be removed and either set pressure tested or replaced with spares which have been previously set pressure tested and stored in accordance with manufacturers recommendations in the specified frequency.

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

The core spray system (CSS), together with the LPCI mode of the RHR system, is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The CSS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the CSS will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Four subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The high pressure coolant injection (HPCI) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HCPI system continues to operate until reactor vessel pressure is below the pressure at which CSS operation or LPCI mode of the RHR system operation maintains core cooling.

The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to deliver greater than or equal to 5600 gpm at reactor pressures between 1182 and 200 psig and is capable of delivering at least 5000 gpm between 1182 and 1205 psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. NPF-85

PECO ENERGY COMPANY

LIMERICK GENERATING STATION, UNIT 2

DOCKET NO. 50-353

1.0 INTRODUCTION

By letter dated January 12, 1999, as supplemented January 29 and March 10, 1999, PECO Energy Company (the licensee) submitted proposed changes to the Limerick Generating Station (LGS), Unit 2, 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 1% to plus or minus 3%. Also, the required number of operable SRVs in operational conditions 1, 2 and 3 will be increased from 11 to 12. The January 29 and March 10, 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.

The licensee requested in its January 29, 1999, letter, that processing of the amendment for LGS, Unit 1 be deferred since the core reload analysis to support implementation of the proposed TS amendment for LGS, Unit 1 will not be completed until April 2000. Hence, this evaluation applies only to LGS, Unit 2.

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 (LTR 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% 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% 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 TS.

(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% tolerance limit.

(e) Evaluation of the plus or minus 3% 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% 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 <u>EVALUATION</u>

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 2 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. Existing TS provides a plus or minus 1% as-found tolerance and plus or minus 1% as-left setpoint tolerance. The proposed modifications would provide a plus or minus 3% as-found tolerance and plus or minus 1% 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% 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. All future reload analyses are expected to assume the revised tolerance.

3.2 Analysis of the Design Basis Overpressurization Event

The licensee is required to reevaluate the limiting design basis pressurization transient using the 3% 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% over design pressure (110% 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% 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 TS. The analysis took credit only for 12 of the 14 SRVs required by the TS. This is acceptable to the staff.

The Limerick 2 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% 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 Limerick. 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 during the upcoming outages for both Units 1 and 2. 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 performance of high pressure systems (pump capacity, discharge pressure, etc.), considering the 3% tolerance limit. Limerick Unit 2 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 system's 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% to 122.1%. 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% setpoint tolerance, the simmer margin would be a minimum of 89.9 psi. The minimum simmer margin using the current plus or minus 1% 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 (MELLL), 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 depressurizes very rapidly and thus, the SRVs are not challenged. Also, the reactor coolant system 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% 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 Limerick Unit 2 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 steam line break outside containment events were evaluated to determine the effects of the increased SRV setpoint tolerance. GE performed 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 2 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% 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% to plus or minus 3%. This is acceptable as described above. In Surveillance Section 4.4.2.2, the following statement has been added: "All safety valves will be recertification tested to meet a ±1% 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 ECCS-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 GE verified that LGS, Unit 2, 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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