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AUTH.NAME AUTHOR AFFILIATION
SHIFFER,J.D. Pacific Gas & Electric Co.
RECIP.NAME RECIPIENT AFFILIATION
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SUBJECT: Forwards application for amend to Licenses DPR-80 & DPR-82, revising Tech specs 3.4.2.1,3.4.2.2 & Table 3.7-2.

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Pacific Gas and Electric Company

77 Beale Street
San Francisco, CA 94106
415/972-7000
TWX 910-372-6587

James D. Shiffer
Vice President
Nuclear Power Generation

September 6, 1989

PG&E Letter No. DCL-89-226

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Re: Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
License Amendment Request 89-11
Revision of Technical Specifications 3.4.2.1, 3.4.2.2, Table 3.7-2, and Associated Bases to Increase Setpoint Tolerances for Safety Valves

Gentlemen:

Enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82. Code safety valves at Diablo Canyon and other plants have incurred numerous surveillance test failures due to setpoint drift. The enclosed license amendment request (LAR) proposes to change the setpoint tolerance of the pressurizer and main steam line code safety valves from ±1 percent to ±3 percent, except for the lowest set pressure bank of main steam line safety valves, which will be changed to -2/+3 percent. These proposed changes are supported by an analysis of the impact of an increased code safety valve tolerance band on plant operations. This proposal is very similar to a change in the main steam safety valve setpoint tolerance recently approved for the Palisades Plant.

Since the changes proposed in this LAR are not required to address an immediate safety concern, PG&E believes that the NRC priority for review and approval of this LAR should be medium. PG&E desires to implement the proposed technical specification changes during the Unit 2 third refueling outage scheduled for February 1990, and therefore requests timely NRC review and approval. PG&E will make the revised technical specifications effective upon receipt of a license amendment from the NRC.

Kindly acknowledge receipt of this material on the enclosed copy of this letter and return it in the enclosed addressed envelope.

Sincerely,

J. D. Shiffer

A001
1/1

cc: J. B. Martin
M. M. Mendonca
P. P. Narbut
H. Rood
P. A. Szalinski
B. H. Vogler
CPUC
Diablo Distribution

Enclosure

2567S/0071K/RLJ/2254

ENCLOSURE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY Diablo Canyon Power Plant Units 1 and 2	Docket No. 50-275 Facility Operating License No. DPR-80 Docket No. 50-323 Facility Operating License No. DPR-82 License Amendment Request No. 89-11

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) hereby applies to amend its Diablo Canyon Power Plant Facility Operating License No. DPR-80 and DPR-82.

The proposed changes amend the Units 1 and 2 Technical Specifications (Appendix A of the Licenses) regarding Technical Specifications 3.4.2.1, 3.4.2.2, Table 3.7-2 and Associated Bases for pressurizer and main steam line safety valves.

Information on the proposed changes is provided in Attachments A, B, and C.

These changes have been reviewed and are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92 or an unreviewed environmental question. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

Subscribed to in San Francisco, California this 6th day of September 1989.

Respectfully submitted,

Pacific Gas and Electric Company

By _____
J. D. Shiffer
Vice President
Nuclear Power Generation

Howard V. Golub
Richard F. Locke
Attorneys for Pacific
Gas and Electric Company

By _____
Richard F. Locke

Subscribed and sworn to before me
this 6th day of September 1989

_____
Adriane D. Tolefree, Notary Public
for the County of Alameda,
State of California

My commission expires December 22, 1992.

2567S/0071K

OFFICIAL SEAL
ADRIANE D. TOLEFREE
NOTARY PUBLIC - CALIFORNIA
ALAMEDA COUNTY
My Comm. Expires Dec. 22, 1992

Attachment A

REVISION OF TECHNICAL SPECIFICATIONS 3.4.2.1, 3.4.2.2, TABLE 3.7-2, AND ASSOCIATED BASES TO INCREASE SETPOINT TOLERANCES FOR SAFETY VALVES

A. DESCRIPTION OF AMENDMENT REQUEST

This license amendment request (LAR) proposes to revise Technical Specifications (TS) 3.4.2.1, 3.4.2.2, and Table 3.7-2 to change the setpoint tolerance for the pressurizer and main steam line code safety valves from ±1 percent to ±3 percent, except for the lowest set pressure bank of main steam line safety valves which will be changed to -2/+3 percent. This LAR also revises TS Bases 3/4.7.1.1 to reference the correct secondary system design pressure and overpressure limit.

The proposed changes to the TS of Operating License Nos. DPR-80 and DPR-82 are noted in the marked-up copy of the applicable TS (Attachment B). Attachment C provides a report summarizing the analysis performed to justify increasing the safety valve setpoint tolerances.

B. BACKGROUND

Overpressure protection of the reactor coolant system (RCS) and main steam system (MSS) is provided in part by the mechanical code safety valves located on the pressurizer and each of four main steam lines. There are a total of three pressurizer code safety valves, set at 2485 psig, and twenty main steam line code safety valves, set at varying setpoints from 1065 to 1115 psig. All these valves are tested just prior to or during refueling outages and must currently meet a tolerance of ±1 percent of the setpoint.

The reactor coolant system and main steam system piping and fittings were designed in accordance with ANSI B31.1. The reactor vessel and pressurizer were designed in accordance with ASME Section III. The main steam system piping is under the jurisdiction of ASME Section I (1968). The safety relief valves were designed in accordance with the requirements of the 1967 Edition through the Summer 1968 Addenda of ASME Section III. The Diablo Canyon Power Plant (DCPP) inservice test (IST) program for safety valves meets the requirements of the 1977 Edition through Summer 1978 Addenda of ASME Section XI.

C. JUSTIFICATION

Setpoint drift has been experienced during the testing of the pressurizer and main steam line safety valves and has been reported in Licensee Event Report (LER) 1-86-018 (PG&E Letter No. DCL-87-030, dated February 23,

1987) and LERs 1-88-018 Rev. 0 (PG&E Letter No. DCL-88-229, dated September 30, 1988) and Rev. 1 (PG&E Letter No. DCL-89-172, dated June 27, 1989). As reported in the above LERs, a significant number of pressurizer and main steam line safety valves have failed to meet the current ±1 percent acceptance criteria on the first lift attempt during surveillance testing. Also as described in the LERs, PG&E has aggressively pursued the cause of the setpoint drift with a comprehensive test program. This program has shown that implementation of an expanded setpoint tolerance range is desireable to assure that the code safety valves stay within the required TS limits and that unnecessary event reporting is eliminated.

Industry operating experience has also demonstrated that a setpoint tolerance for the pressurizer and main steam line safety valves of only ±1 percent results in safety valves frequently failing surveillance tests. IE Information Notice 86-56, "Reliability of Main Steam Safety Valves," tabulated the number of problems with setpoint drift as reported in various LERs. This information notice discussed the various safety concerns that may be encountered with setpoint drift. PG&E has considered these concerns in the proposed relaxed tolerance limits. Other plants, such as Trojan, Palisades, Yankee Rowe, Calvert Cliffs, and LaSalle, have been granted setpoint tolerance relaxations.

The ASME Section XI schedule used previously for testing additional valves will continue to be used, except that the criteria for testing additional valves will be ±3 percent of the setpoint instead of ±1 percent. However, all valves that are tested and found outside the ±1 percent band will be reset to within ±1 percent.

PG&E has reviewed the TS Bases 3/4.7.1.1 and proposes to revise the secondary system design pressure and the overpressure limit. The correct value for the secondary system design pressure is 1085 psig per the Westinghouse equipment specifications. The overpressure limit of 110 percent of the design pressure is consistent with the applicable sections of the ASME code and all necessary components have been verified to meet this limit. The 110 percent value is also consistent with the Standard Technical Specifications.

D. SAFETY EVALUATION

As shown below, the proposed tolerance relaxation meets overpressure protection requirements and all ASME code requirements. In addition, evaluation of the FSAR Update accident analyses affected by the change has demonstrated that the conclusions for these analyses as stated in the FSAR Update remain valid, and results in no additional operational concerns. All evaluations were conservatively performed assuming the use of VANTAGE 5 fuel.

Overpressure Protection

There are five FSAR Update Condition II accidents which result in significant RCS and MSS pressure increases. The accidents are (1) an uncontrolled rod withdrawal from full power, (2) loss of reactor coolant

flow, (3) loss of external electrical load/turbine trip, (4) loss of normal feedwater, and (5) loss of all AC power to the station auxiliaries. Safety valve actuation is only required to limit the system pressure upon loss of external electrical load, loss of normal feedwater, and loss of all AC power. Of these, the loss of external electrical load/turbine trip is the most limiting. Using assumptions consistent with those in the DCPP FSAR Update and NUREG-0800, "Standard Review Plan," PG&E analyzed the loss of external electrical load/turbine trip using an NRC-approved version of the RETRAN computer code. PG&E's RETRAN model was benchmarked against plant data obtained from a DCPP Unit 2 startup test where the main turbine was manually tripped from 100 percent power. This RETRAN model was also used to reanalyze the most limiting of the four loss of load/turbine trip events presented in the DCPP FSAR Update. The RETRAN model analysis showed good agreement with the FSAR Update and startup test turbine trip results.

Three cases were considered in the analysis. As shown in Attachment C, the cases correspond to primary and secondary safety valve setpoints equal to 2, 3 and 3.5 percent above nominal setpoint (see Table 1 for case summary). In all cases, the pressurizer safety valves were assumed to open linearly (with pressure) over a 3 percent pressure range above the setpoint. In the case where the setpoint is 2 percent above nominal, the pressurizer safety valves are modeled as starting to open at 2 percent above nominal setpoint and being fully open at 5 percent above nominal setpoint. All twenty main steam line safety valves were assumed to open at a setpoint equal to 2, 3 and 3.5 percent above the highest of all the nominal setpoints, and were assumed to open instantaneously once they reached that setpoint. The peak pressurizer pressure for a +2 percent setpoint deviation is 2599 psia. The peak RCS pressure was 2708 psia, which is 51 psi higher than the existing FSAR Update results but still well within the RCS design pressure limit of 2750 psia. This peak pressure occurs in the lower plenum of the reactor vessel. The peak main steam pressure for this case is 1154 psia, which is only about 21 psi higher than the FSAR Update case, and is also well within the acceptance criteria of 1210 psia. The safety valve setpoint increase has only minor effects on the pressurizer water level, RCS average temperature, and secondary steam temperature.

The RCS peak pressure increase for a +3 percent setpoint deviation is about 75 psi above the FSAR Update value, or 2732 psia. The peak RCS pressures and main steam pressure are within the acceptance limits (110 percent of design value).

If the setpoints for both the pressurizer and steam line safety valves are increased to +3.5 percent above their nominal setpoint, the RCS peak pressure will reach 2743 psia, which is about 7 psi lower than the acceptance limit. The main steam pressure will not exceed the 110 percent design value of the steam generator vessel.

Therefore, based on the loss of external electrical load/turbine trip analysis, the safety valve tolerance limit for both the pressurizer and steam lines can be changed from +1 percent to +3.5 percent.

TABLE 1

PRESSURIZER AND MAIN STEAM LINE SAFETY VALVE OPENING SETPOINTS
(PSIG)

Case	Percent Above Nominal	Pressurizer Safety Valves Begin To Open	Pressurizer Safety Valves Fully Open	Main Steam Safety Valves Open
1	2%	2535	2609	1137
2	3%	2560	2634	1148
3	3.5%	2572	2647	1154

This RETRAN model was independently reviewed by EI International, Inc. In addition, the assumptions and results of the PG&E overpressure analysis were reviewed by Westinghouse and by Cermak Fletcher & Associates, Inc. The results of the RETRAN analysis are presented in Attachment C.

It is concluded that with respect to overpressure protection, the safety valve tolerance limit for both the pressurizer and main steam line may be relaxed to +3 percent.

ASME Code Requirements

The Diablo Canyon main steam line and pressurizer safety valves were procured and constructed to specifications and design codes which did not specify tolerances for the set pressure. Subsequent to this (and in agreement with then-current construction Code requirements), the Diablo Canyon Technical Specifications were written with a tolerance on the valve set pressures of ±1 percent.

Recent editions of Section XI (beginning with the Winter 1985 Addenda) have endorsed ANSI/ASME OM-1-1981 requirements for IST of such safety valves. OM-1-1981 acknowledges that a drift in the setpoint of up to 3 percent would not be considered an unreasonable deviation from the stamped set pressure. It is PG&E's position that a setpoint tolerance of +3 percent is consistent with the requirements of OM-1-1981 and is compatible with the Code requirements of the present DCPP IST program and valve design documents.

Non-LOCA Accident Evaluation

Many of the non-LOCA accidents are not affected by the assumption of minimum and maximum tolerance on either the main steam line or pressurizer safety valves. This is because the pressures reached in the transient never approach the values for the lift setpoints. The pressure in a given transient would have to reach 2425 psia on the primary side or 1096 psia on the secondary side for the transient to begin to be affected by the increased tolerance, where the 1096 psia secondary setpoint reflects the Westinghouse transient methodology of modeling all main steam line safety valves as lifting at the highest of the five variable setpoints. These are the lowest lift setpoints for the safety valves for a -3 percent tolerance. All FSAR Update Chapter 15 non-LOCA transients have been evaluated and most are not significantly affected by the increased pressurizer and main steam line safety valve increased setpoint tolerance. The following accidents could possibly be impacted either because of Departure from Nucleate Boiling Ratio (DNBR) concerns or other licensing-basis acceptance criteria:

- loss of external load/turbine trip
- loss of normal feedwater/station blackout
- feedwater line break

- partial loss of forced reactor coolant flow
- complete loss of forced reactor coolant flow
- locked rotor/shaft break
- uncontrolled rod cluster control assembly bank withdrawal at power
- startup of an inactive loop

RETRAN was used to reanalyze the loss of external load/turbine trip; this analysis showed that the design pressure was not exceeded with an expanded pressurizer and main steam line safety valve setpoint tolerance band. The locked rotor/shaft break accident was reviewed and it was found that this event only uses a small fraction of the available safety relief capacity for pressure relief. The peak pressure will remain below the design limit for the proposed change and the conclusions in the FSAR Update and the VANTAGE 5 fuel analyses remain valid.

The quantity of steam mass released from the main steam line safety valves will not be significantly affected by the revised tolerance values. The primary effect of the increase in the safety valve tolerance on the impacted accidents is to alter the time at which the valves would open during the pressure increase transient. Even though the initial steam flow through the safety valves will increase due to a higher back pressure, the integrated mass release relieved from the secondary side is not expected to be significantly increased.

The impacted non-LOCA accidents would have slightly lower DNBR values if the primary safety valves lift at lower initial pressures. However, Westinghouse has judged the accidents which are DNBR criterion events to have a negligible decrease in DNBR. The main steam safety valve tolerance relaxation has even a lesser impact on DNBR. The DNBR transient is not changed significantly for the relaxation in setpoint tolerance and the conclusions reached for the DNBR accidents in the FSAR Update and the VANTAGE 5 fuel analyses remain valid.

There is significant margin between the transient hot leg temperature and the hot leg saturation temperature in the existing analyses for the feedwater line break. If the pressurizer safety valves open 3 percent below nominal, the saturation pressure may be reduced slightly, but significant margin remains. The conclusions previously reported in the FSAR Update and the VANTAGE 5 fuel analyses remain valid.

The locked rotor and rod ejection events will not be affected by the setpoint tolerance as it relates to the peak heat flux calculation. These analyses assume a conservative pressure which is less than the steady state RCS pressure. This pressure is maintained at a constant value throughout the transient so as to maximize the peak heat flux. Therefore, the FSAR Update and VANTAGE 5 fuel analyses conclusions remain valid.

The effect of the setpoint tolerance change on long-term heat removal for both the loss of normal feedwater and station blackout events was considered. The competing effects of increased/decreased heat removal capability on the secondary side and higher/lower average coolant temperatures on the primary side result in no significant impact on the heat removal capability of the secondary system. There is significant margin in these events to filling the pressurizer and the conclusions presented in the FSAR Update and the VANTAGE 5 fuel analyses remain valid.

LOCA Accident Evaluation

A safety evaluation for a safety valve setpoint tolerance relaxation to ±3 percent on the Chapter 15 FSAR Update LOCA accidents was performed in order to justify relaxation of the current ±1 percent tolerance on the pressurizer and main steam line safety valves. PG&E has taken credit for the steam generator power operated relief valves (SGPORVs).

The current large break LOCA analysis forming the licensing basis for Diablo Canyon Units 1 and 2 (break size greater than or equal to 1.0 sq. ft.) results in a very rapid (approximately 30 seconds) depressurization of the RCS from the operating pressure to a pressure slightly above that of the containment. Because of the rapid primary depressurization, the secondary side of the steam generators quickly becomes a heat source rather than a heat sink such that the main steam safety valves and SGPORVs are not challenged. The pressurizer safety valves are also not challenged because of the RCS depressurization. Therefore, the proposed safety valve tolerance relaxation will have no effect on the large break LOCA analysis.

The current small break LOCA analysis forming the licensing basis for Diablo Canyon Units 1 and 2 causes the system to depressurize to a pressure slightly above that of the steam generator secondary side pressure relief. The main steam safety valves provide a significant path for RCS energy release until steam venting through the break occurs. The primary pressure and the duration of time that the RCS primary remains above the secondary side pressure is governed by the rate of decay energy removal through the break and the amount of heat transferred to the steam generator secondary side. A slight increase in RCS pressure is computed to occur during this portion of the transient due to the higher secondary pressure as a result of the relaxed tolerances. Analysis has shown that increasing the secondary pressure, and as a result, the RCS pressure, results in higher peak cladding temperatures (PCT). The current Diablo Canyon small break analysis assumes nominal main steam line safety valve setpoints. An evaluation of PCT shows there is still a significant margin to the 2200°F regulatory limit. Further, since the small break LOCA is a depressurization transient, the pressurizer safety valves are not challenged. Conversely, decreasing the secondary pressure results in a PCT benefit which has not been assessed. Therefore, the relaxed pressurizer and main steam line safety valve tolerance with SGPORV operation is acceptable with respect to the small break LOCA analysis.

Post-LOCA hot leg recirculation switchover time is determined for inclusion in emergency procedures to ensure long term core cooling by precluding boron precipitation in the reactor vessel following boiling in the core. This time is dependent on power level as well as the RCS, refueling water storage tank, and accumulator water masses and boron concentrations. Changes in the steam generator secondary pressure influence the RCS pressure and masses assumed in the switchover analysis. The effect of the proposed main steam line safety valve tolerance relaxation with SGPORV operation has been evaluated and is insignificant. Further, operation of the pressurizer safety valves will not affect the analysis since calculations are based on total mass inventories.

The LOCA hydraulic forcing functions acting upon the vessel, internals, and loop are a function of the primary system geometry and primary operating conditions. The peak forces are generated within the first seconds after break initiation. For this reason, the forces model does not consider the pressurizer safety valves or the effects of the secondary side. As such, the relaxation in pressurizer and main steam line safety valve lift set tolerances to +3 percent with the SGPORVs will have no effect on the magnitude or frequency of the LOCA hydraulic forcing functions provided in the Diablo Canyon FSAR Update.

Post-LOCA long-term core cooling will not be affected by an increase in pressurizer and main steam line safety valves setpoint tolerances because no change in the sump boron concentration would occur. Sump boron concentration is determined by the accumulation of all potential water sources in the containment, based on each respective source boron concentration. A scenario has not been envisioned whereby changes in safety valve or SGPORV operation would result in spilling additional non-borated water, reduce the inventory of borated water, or limit component boron concentration as used in the mass average calculation used in the evaluation. It is concluded that there would be no change to the long-term cooling capability of the emergency core cooling system as a result of increased valve tolerances with SGPORV operation.

Steam Generator Tube Rupture Accident Evaluation

The FSAR Update Steam Generator Tube Rupture (SGTR) analysis is performed to evaluate the radiological consequences of an SGTR accident. The major factors that affect the resultant offsite doses are the amount of radioactivity in the reactor coolant, the total amount of primary coolant transferred to the secondary side of the ruptured steam generator through the ruptured tube, and the steam released from the ruptured steam generator to the atmosphere. The amount of radioactivity in the reactor coolant

assumed in the FSAR Update SGTR analysis is not affected by the changes in the pressurizer or main steam line safety valve setpoint tolerances. The effect of relaxing the pressurizer safety valve setpoint tolerances to ±3 percent, the lowest main steam line safety valve setpoint tolerances to -2/+3 percent and the remaining main steam line safety valve setpoint tolerances to ±3 percent on the FSAR Update SGTR analysis and the updated SGTR analysis has been addressed by a Westinghouse evaluation.

Since the pressurizer pressure decreases following an FSAR Update SGTR event, the pressurizer safety valve will not actuate. Thus, the increased pressurizer safety valve setpoint tolerances would not affect the FSAR Update SGTR analysis. The FSAR Update SGTR analysis for Diablo Canyon Units 1 and 2 assumes that the primary to secondary break flow is terminated at 30 minutes after initiation of the SGTR event. A decrease in the safety valve setpoint pressure will increase the calculated equilibrium break flow rate, and vice versa. Since an increase in the setpoint tolerance in the positive direction will reduce the calculated break flow and offsite doses, it is only required to evaluate the effect of the increase in the setpoint tolerance in the negative direction. The change in the negative direction would result in a change of the assumed lowest safety valve setpoint from 1065 psig to 1043 psig. The effect of a decrease in the assumed lowest safety valve setpoint from 1065 psig to 1043 psig is estimated to be an increase of approximately 0.8 percent in the integrated primary to secondary leakage, and an increase of approximately 0.3 percent in the steam released from the ruptured steam generator.

The thyroid and whole body doses are estimated to increase by approximately 1 percent for a decrease in the assumed lowest safety valve setpoint from 1065 psig to 1043 psig. The resultant offsite doses with the effect of the increased pressurizer and main steam line safety valves setpoint tolerances, while greater than currently reported in the FSAR Update, do not constitute an increase in the consequence of the accident. This judgement is based on the fact that the resultant doses are still well within the acceptable exposure criteria.

These conclusions are valid for the current FSAR Update SGTR analysis as well as the revised SGTR analysis for Diablo Canyon currently under review by the NRC Staff.

Conclusion

The non-LOCA safety evaluation supports the conclusion that the proposed tolerance of +3 percent for the pressurizer and main steam line safety valves is acceptable. This conclusion is reached based on the assessment that the non-LOCA acceptance criteria are met for this increase in opening setpoints for all the non-LOCA accidents. It is concluded that the tolerances on the pressurizer and main steam line safety valves as given in the Technical Specifications for Diablo Canyon may be changed to specify ±3 percent. This conclusion applies to both Diablo Canyon units for the current fuel as well as the planned VANTAGE 5 fuel.

The LOCA safety evaluation supports the conclusion that the effects of increasing pressurizer and main steam line safety valve tolerances to ±3 percent (taking credit for the SGPORV as described) at Diablo Canyon Units 1 and 2 do not result in exceeding any design or regulatory limit. Therefore, it is concluded that the proposed Technical Specification valve tolerance relaxation is acceptable.

The increase in safety valve setpoint tolerance from ±1 percent to ±3 percent (with the exception of the main steam line safety valves having the lowest setpoint which were assumed to have their setpoint tolerance relaxed by -2/+3 percent) has been evaluated with respect to both LOCA and non-LOCA events for impact on the radiological consequences of accidents. There is no impact on the radiological consequences of any accident except for the SGTR, and the impact of the proposed change on the SGTR doses is small and does not affect the acceptability of the doses.

Operational Concerns

The DCPP operating pressure is 2250 psia and the high pressurizer pressure reactor trip is 2400 psia. The existing tolerance (±1 percent) on the safety valve setpoint pressure is 2475 psia to 2525 psia. The previously discussed safety evaluation justifies relaxation to ±3 percent, or 2425 psia to 2575 psia.

Operation with safety valves at -3 percent tolerance would decrease the margin to normal operating pressure from 225 psi to 175 psi. This has been evaluated during testing at an offsite testing facility and is not judged to be a problem in terms of valve leakage, simmering, or weeping. Also, operation with the valves at a tolerance of -3 percent would place the reactor trip pressure 25 psi below the -3 percent setpoint of 2425 psia. This is not considered to be a problem because the uncertainty on the high pressurizer pressure setpoint is less than 25 psi. Also, all valves tested will continue to be reset to within ±1 percent.

A similar evaluation was performed for the main steam line safety valves and the same conclusion was reached.

E. NO SIGNIFICANT HAZARDS EVALUATION

PG&E has evaluated the hazard considerations involved with the proposed amendment focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The following evaluation is provided for the three categories of the significant hazards consideration standards.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The pressurizer and main steam line safety valves are designed to mitigate transients by preventing overpressurization of the RCS and MSS. The proposed change does not alter this design basis. The revised analysis shows that the probability or consequences of all previously analyzed accidents are not changed by increasing the setpoint tolerance of the safety valves.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   There is no physical alteration to any plant system, nor is there a change in the method in which any safety related system performs its function. Any pressurizer safety valve lifting at the extremes of the proposed tolerance will not result in a low lift setpoint that is less than the pressurizer high pressure reactor trip or a high lift setpoint that allows RCS overpressurization. Any main steam safety valve lifting at the extremes of the proposed tolerance will not result in a low lift setpoint that is less than the normal "no load" system pressure or a high lift setpoint that allows MSS overpressurization.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

   With the increased setpoint tolerances, the pressurizer and main steam line safety valves will still prevent pressure from exceeding 110 percent of design pressure in accordance with the ASME code. All conclusions for the FSAR Update accident analyses are not affected by the change and remain valid.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

In conclusion, based on the above evaluation, PG&E submits that the activities associated with this LAR satisfy the no significant hazards consideration standards of 10 CFR 50.92(C) and, accordingly, a no significant hazards consideration finding is justified.

G. ENVIRONMENTAL EVALUATION

PG&E has evaluated the proposed change and determined that the change does not involve (i) a significant hazards consideration (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released off site or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(B), an environmental assessment of the proposed change is not required.