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AUTH.NAME AUTHOR AFFILIATION
FUJIMOT,W.H. Pacific Gas & Electric Co.
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Pacific Gas and Electric Company

Nuclear Technical Services, A10G
333 Market Street, Room 8024
P.O. Box 770000
San Francisco, CA 94177
415/973-0600
Fax 415/973-1427

Warren H. Fujimoto
Vice President

August 5, 1993

PG&E Letter No. DCL-93-198

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
Response to NRC Generic Letter 93-04

Gentlemen:

PG&E's response to NRC Generic Letter (GL) 93-04, "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10 CFR 50.54(f)," dated June 21, 1993, is provided in Enclosure 1. A summary of the preliminary results of the generic safety analysis program conducted by the Westinghouse Owners Group (WOG) and reviewed by PG&E for its applicability to Diablo Canyon Units 1 and 2 is provided in Enclosure 2. PG&E has implemented the NRC-recommended compensatory actions listed in GL 93-04 that were originally included in a Westinghouse Nuclear Safety Advisory Letter 93-007, dated June 11, 1993. These actions are discussed in Enclosure 3.

The generic letter requires that, within 45 days from the date of the generic letter, licensees shall provide an assessment of whether or not the licensing basis for each facility is still satisfied with regard to the requirements for system response to a single failure in the rod control system (General Design Criterion 25 or equivalent). If the assessment (Required Response 1.(a) in GL 93-04) indicates that the licensing basis is not satisfied, then the licensee must describe compensatory short-term actions consistent with the guidelines contained in GL 93-04 (Required Response 1.(b)), and within 90 days, provide a plan and schedule for long-term resolution (Required Response 2.). Subsequent correspondence between the WOG and the NRC resulted in schedular relief for Required Response 1.(a) and the first part of 1.(b) (NRC letter to Roger Newton of the WOG dated July 26, 1993). This portion of the required actions will now be included with the 90-day response.

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This letter summarizes the compensatory actions taken by PG&E and the review of the preliminary results of the WOG generic safety analysis program in response to the Salem rod control system failure event. PG&E considers these actions to be complete with respect to the 45-day required responses to GL 93-04, as amended by the July 26 NRC letter to Roger Newton.

Sincerely,

Warren Fujimoto

Warren H. Fujimoto

Subscribed and sworn to before me this 5th day of August 1993.

Wendy Sproul

Wendy Sproul, Notary Public

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

Richard F. Locke

Richard F. Locke

cc: Bobby H. Faulkenberry
Ann P. Hodgdon
Mary H. Miller
Sheri R. Peterson
CPUC
Diablo Distribution

OFFICIAL NOTARY SEAL
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Enclosures

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ENCLOSURE 1

RESPONSE TO NRC GENERIC LETTER 93-04

Background

On May 27, 1993, operators at the Salem Nuclear Generating Station, Unit 2, experienced problems with the rod control system. During an attempt to withdraw Shutdown Bank A, the operator observed that the analog rod position indicator (ARPI) did not indicate that the control rods were being withdrawn. The operator stopped attempting to withdraw rods at 20 steps as indicated on the group demand indicator. At this time, the ARPI indicated that all of the rods in Shutdown Bank A were at the 0 step position. The operator then attempted to insert Shutdown Bank A.

However, one control rod (1SA3) withdrew to 8 steps as indicated by the ARPI while the group demand indicator counted down from 20 steps to 6 steps. The operator continued to try to insert the Shutdown Bank A control rods until the group demand indicator showed a rod position of zero. The operator observed that the indicated position on the ARPI for control rod 1SA3 was 15 steps. Technicians then removed power from the rod by pulling fuses, and rod 1SA3 dropped to the 0 step position as indicated by the ARPI.

All Westinghouse-designed, pressurized-water reactors (PWRs), except Haddam Neck, use the rod control system installed at Salem Unit 2. Initial assessments by Westinghouse and testing by PG&E, preliminarily determined that a single failure in the rod control system could result in unintended rod withdrawal movements of control rods. Although the reactor protection system is independent of the rod control system logic and, therefore, the reactor trip function is not compromised, there remains a concern that a previously unanticipated single-failure mechanism may exist in the control system that can initiate or aggravate reactivity excursions and result in fuel failure.

PG&E's response to Generic Letter (GL) 93-04, "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10 CFR 50.54(f)," is provided below. The response is consistent with the schedular relief granted in an NRC letter (A. C. Thadani) to the Westinghouse Owners Group (WOG) (Roger Newton) on July 26, 1993.

"1. Within 45 days from the date of this generic letter:

(a) Provide an assessment of whether or not the licensing basis for each facility is still satisfied with regard to the requirements for system response to a single failure in the rod control system and provide a supporting discussion for this assessment in light of the information generated as a result of the Salem event."

PG&E RESPONSE

Based on the schedular relief granted by the NRC to the WOG on July 26, 1993, PG&E will address Required Response 1.(a) within 90 days of the date of issuance of GL 93-04.

> *"1. (b) If the assessment in 1(a) indicates that the licensing basis is not satisfied*
>
> - *provide an assessment of the impact of potential single failures in the rod control system on the licensing basis of the facility"*

PG&E RESPONSE

Based on the schedular relief granted by the NRC to the WOG on July 26, 1993, PG&E will address the first part of Required Response 1.(b) within 90 days of the date of issuance of GL 93-04.

> *"1. (b)*
> - *describe any compensatory short-term actions taken or that will be taken to address any actual or potential degraded or nonconforming conditions (see Generic Letter 91-18, Reference 1) such as*
>   - *additional cautions or modifications to surveillance and preventive maintenance procedures*
>   - *additional administrative controls for plant startup and power operation*
>   - *additional instructions and training to heighten operator awareness of potential rod control system failures and to guide operator response in the event of a rod control system malfunction"*

PG&E RESPONSE

The following response to the second part of Required Response 1.(b) includes information reviewed by PG&E that was received from the WOG in Letter OG-93-53 issued to the WOG primary representatives on July 30, 1993.

> *" - additional cautions or modifications to surveillance and preventive maintenance procedures"*

PG&E has reviewed the surveillance procedures and preventive maintenance practices at Diablo Canyon Power Plant (DCPP) Units 1 and 2 and ensured that they adequately (1) include the steps necessary to ensure rod control system

operability, and (2) incorporate maintenance troubleshooting. Therefore, preventive maintenance and surveillance procedures require no modifications to testing frequencies, scope, or additional cautions.

In addition, PG&E has implemented the NRC-recommended compensatory actions listed in GL 93-04 (issued in a Westinghouse Nuclear Safety Advisory Letter (NSAL) 93-007, dated June 11, 1993); see Enclosure 3.

Westinghouse did not make any initial recommendations regarding surveillance or preventive maintenance procedures. Based on the response provided in WOG Letter OG-93-42, dated July 2, 1993, there was no perceived need to increase the frequency of testing on a permanent or generic basis. Public Service Electric & Gas Company (PSE&G) had committed to a temporary increase in testing, but only until it was demonstrated that the rod control system was operating properly and with confidence. Upon further review, Westinghouse recommended that utilities ensure their surveillance testing will demonstrate rod control system operability and address maintenance trouble-shooting. The PG&E actions described above satisfy the Westinghouse recommendations.

> *" - additional administrative controls for plant startup and power operation"*

Based on PG&E's review of the DCPP abnormal operating procedures and annunciator response procedures, no additional administrative controls for plant startup and power operations are needed.

PG&E has also implemented the NRC-recommended compensatory actions listed in GL 93-04 (issued in the Westinghouse NSAL 93-007); see Enclosure 3.

Following the rod withdrawal event at Salem Unit 2, PSE&G committed the Salem units to startup by dilution. Neither Westinghouse nor the WOG has endorsed this requirement. In actual operation, the operators would be aware of abnormal rod movement and terminate rod demand prior to ever reaching criticality. The operator would be manually controlling the rod withdrawal such that rod misstepping would be detected in less than one minute. Thus, the WOG and Westinghouse have concluded that startup by dilution is not required in response to the Salem rod control system failure event.

> *" - additional instructions and training to heighten operator awareness of potential rod control system failures and to guide operator response in the event of a rod control system malfunction"*

Both Westinghouse and the WOG have, at various times, recommended that licensees provide additional discussion, training, standing orders, etc. to ensure that their operators are aware of what transpired at Salem. To enhance operator awareness, a Shift Order, dated June 22, 1993, was issued to all DCPP plant control operators advising them of the Salem event and the possibility for a similar failure at DCPP. Additionally, PG&E has implemented the NRC-recommended compensatory actions listed in GL 93-04; see Enclosure 3.
PG&E has reviewed the event and has concluded that no additional compensatory measures, other than administrative procedures already in place and adherence to Technical Specification requirements, are required.

"2. *If the assessment in 1(a) indicates that the licensing basis is not satisfied, within 90 days from the date of this generic letter provide a plan and schedule for the long-term resolution of this issue."*

PG&E RESPONSE

PG&E will address Required Response 2. within the specified 90-day requirement established from the date of issuance of GL 93-04.

ENCLOSURE 2

SUMMARY OF THE GENERIC SAFETY ANALYSIS PROGRAM

Introduction

As part of the Westinghouse Owners Group (WOG) initiative, the WOG Analysis Subcommittee is working on a generic approach to evaluate for all Westinghouse plants the safety significance of an asymmetric rod cluster control assembly (RCCA) withdrawal. The purpose of the program is to analyze a series of asymmetric rod withdrawal cases from both subcritical and power conditions to demonstrate that departure from nucleate boiling (DNB) does not occur.

The generic information provided in this enclosure was received from the WOG (Letter OG-93-53, dated July 30, 1993) and determined to be applicable to Diablo Canyon Power Plant (DCPP) Units 1 and 2. PG&E considers this information preliminary and will continue to follow the WOG industry-sponsored efforts to resolve the rod control system issue, which include: (1) the rod control evaluation program to assess the historical performance of the rod control system, and to determine the type of rod motion that can occur when the drive mechanisms are subjected to corrupted current orders; and (2) the safety analysis program to show compliance with General Design Criterion 25.

The current Westinghouse analysis methodology for a bank withdrawal at power and from subcritical uses point-kinetics and one-dimensional kinetics transient models, respectively. These models use conservative constant reactivity feedback assumptions that result in an overly conservative prediction of the core response for these events.

A three-dimensional spatial kinetics/systems transient code (LOFT/SPNOVA) is being used to show that the localized power peaking is not as severe as current codes predict. The three-dimensional transient analysis approach uses a representative standard 4-loop Westinghouse plant with conservative reactivity assumptions. Limiting asymmetric rod withdrawal statepoints (i.e., conditions associated with the limiting time in the transient) are established for the representative plant, which can be applied to all Westinghouse plants. Differences in plant designs are addressed by using conservative adjustment factors to make a plant-specific DNB assessment.

Description of Asymmetric Rod Withdrawal

The accidental withdrawal of one or more RCCAs from the core is assumed to occur and results in an increase in the core power level and the reactor coolant temperature and pressure. If the reactivity worth of the withdrawn rods is sufficient, the reactor power and/or temperature may increase to the point where the transient is automatically terminated by a reactor trip on a High Nuclear Flux or Over-Temperature Delta-T (OTDT) protection signal. If the reactivity increase is small, the reactor power will reach a peak value and then decrease due to the negative feedback effect caused by the moderator

temperature rise. The accidental withdrawal of a bank or banks of RCCAs in the normal overlap mode is a transient that is specifically considered in plant safety analysis reports. The consequences of a bank withdrawal accident meet Condition II criteria (no DNB). If, however, it is assumed that less than a full group or bank of control rods is withdrawn, and these rods are not symmetrically located around the core, this can cause a "tilt" in the core radial power distribution. The "tilt" could result in a radial power distribution peaking factor that is more severe than is normally considered in the plant safety analysis report and, therefore, could cause a loss of DNB margin.

Due to the imperfect mixing of the fluid exiting the core before it enters the hot legs of the reactor coolant loops, there can be an imbalance in the loop temperatures and, therefore, in the measured values of T-avg and delta-T, which are used in the OTDT protection system for the core. The radial power "tilt" may also affect the ex-core detector signals used for the High Nuclear Flux trip. The axial offset in the region of the core where the rods are withdrawn may become more positive than the remainder of the core, which can result in an additional DNB penalty.

Methods

The LOFT5 computer code is used to calculate the plant transient response to an asymmetric rod withdrawal. The LOFT5 code is a combination of an advanced version of the LOFT4 code (Reference 1), which has been used for many years by Westinghouse in the analysis of the reactor coolant system (RCS) behavior to plant transients and accidents, and the advanced nodal code SPONOVA (Reference 2).

LOFT5 uses a full-core model, consisting of 193 fuel assemblies with one node per assembly radially and 20 axial nodes. Several "hot" rods are specified with different input multipliers on the hot rod powers to simulate the effect of plants with different initial $F\Delta H$ values. ($F\Delta H$, the nuclear enthalpy rise hot channel factor, is defined as the ratio of the integral of linear power along the fuel rod with the highest integrated power to the average rod power.) A "hot" rod represents the fuel rod with the highest $F\Delta H$ in the assembly, and is calculated by SPNOVA within LOFT5. DNB ratios are calculated for each hot rod within LOFT5 with a simplified DNB-evaluation model using the WRB-1 correlation. The DNB ratios resulting from the LOFT5 calculations are used for comparison purposes.

A more detailed DNB ratio analysis is performed at the limiting transient statepoints from LOFT5 using the THINC-IV code (Reference 3) and the Revised Thermal Design Procedure (RTDP). The RTDP applies to all Westinghouse plants for this specific analysis, maximizes DNB ratio margins, is approved by the NRC, and is licensed for a number of Westinghouse plants. The LOFT5-calculated DNB ratios are conservatively low when compared to the THINC-IV code results.

Assumptions

The initial power levels chosen for the performance of bank and multiple RCCA withdrawal cases are 100, 60, and 10 percent and hot zero power (HZP). These power levels are the same powers considered in the RCCA bank withdrawal at power and bank withdrawal from subcritical events presented in the plant Safety Analysis Reports. The plant, in accordance with the RTDP, is assumed to be operating at nominal conditions for each power level examined. Therefore, uncertainties will not affect the results of the LOFT5 transient analyses. For the at-power cases, all reactor coolant pumps are assumed to be in operation. For the HZP case (subcritical event), only two of four reactor coolant pumps are assumed to be in operation. A "poor mixing" assumption is used for the reactor vessel inlet and outlet mixing model.

Results

A review of the results presented in Reference 4 indicates that for asymmetric rod withdrawal cases analyzed with the LOFT5 code, the DNB design basis is met. As demonstrated by the adjustment factor (A-Factor) approach (described below) for addressing various combinations of asymmetric rod withdrawals, the single most-limiting case is plant-specific and is a function of rod insertion limits, rod control pattern, and core design. The results of the A-Factor approach also demonstrate that the cases analyzed with the LOFT5 computer code are sufficiently conservative for a wide range of plant configurations for various asymmetric rod withdrawals. In addition, when the design $F_{\Delta H}$ is taken into account on the representative plant, the DNB ratio criterion is met for the at-power cases.

At HZP, a worst-case scenario (three rods withdrawn from three different banks, which is not possible) shows a non-limiting DNB ratio. This result is applicable to all Westinghouse plants.

Plant Applicability

The three-dimensional transient analysis approach uses a representative standard 4-loop Westinghouse plant with bounding reactivity assumptions with respect to the core design. This results in conservative asymmetric rod(s) withdrawal statepoints for the various asymmetric rod withdrawals analyzed. The majority of the cases analyzed either did not generate a reactor trip or were terminated by a High Neutron Flux reactor trip. For the OTDT reactor trip, no credit is assumed for the $f(\Delta I)$ penalty function. ($f(\Delta I)$ is a function of the indicated difference between the power indicated by the top and bottom detectors of the power range nuclear ion chambers.) The $f(\Delta I)$ penalty function reduces the OTDT setpoint for highly skewed positive or negative axial power shapes. Compared to the plant-specific OTDT setpoints, including credit for the $f(\Delta I)$ penalty function, the setpoint used in the LOFT5 analyses is conservative; i.e., for those cases that tripped on OTDT, a plant-specific OTDT setpoint with the $f(\Delta I)$ penalty function will result in an earlier reactor trip than the LOFT5 setpoint. This ensures that the statepoints generated for those cases that trip on OTDT are conservative for all Westinghouse plants.

With respect to the neutronic analyses, an A-factor was calculated for a wide range of plant types and rod control configurations. The A-factor is defined as the ratio between the design $F_{\Delta H}$ and the change in the maximum transient $F_{\Delta H}$ from the symmetric and asymmetric RCCA withdrawal cases. An appropriate and conservative plant-specific A-factor was calculated and used to determine the corresponding DNB ratio penalty or benefit. With respect to the thermal-hydraulic analyses, differences in plant conditions (including power level,

RCS temperature, pressure, and flow) are addressed by sensitivities performed using the THINC-IV code. These sensitivities are used to determine additional DNB ratio penalties or benefits. Uncertainties in the initial conditions are accounted for in the DNB design limit. Once the differences in plant design were accounted for by the adjustment approach, plant-specific DNB ratio calculations can be generated for all Westinghouse plants.

Conclusions

Using the approach of the WOG, the generic analyses applied to Diablo Canyon Units 1 and 2 demonstrate that DNB does not occur for the worst-case asymmetric rod withdrawal.

References

1. Burnett, T. W. T. et al., LOFTRAN Code Description, WCAP-7907-A, April 1984

2. Chao, Y. A. et al., SPNOVA - A Multi-Dimensional Static and Transient Computer Program for PWR Core Analysis, WCAP-12394, September 1989

3. Friedland, A. J. and S. Ray, Improved THINC IV Modeling for PWR Core Design, WCAP-12330-P, August 1989

4. Huegel, D. et al., Generic Assessment of Asymmetric Rod Cluster Control Assembly Withdrawal, WCAP-13803, August 1993

ENCLOSURE 3

RESPONSE TO NUCLEAR SAFETY ADVISORY LETTER 93-007

Generic Letter 93-04 included recommendations by Westinghouse that the NRC Staff judged to be prudent actions. These recommendations and PG&E responses are provided below.

"Westinghouse issued a Nuclear Safety Advisory Letter (NSAL) 93-007, dated June 11, 1993, recommending the following actions:

1. *Licensed operators should continue the normal process of verifying that rod motion is proper for required movement.*

2. *Licensees should confirm the functionality of rod deviation alarms.*

3. *Operators should review the advisory letter to ensure their understanding of the event.*

4. *The Westinghouse Owners Group (WOG) survey its members regarding rod misalignment events and provide a summary.*

Implementation of the recommendations in the Westinghouse NSAL is judged by the NRC staff to be a prudent action."

PG&E Response to Item 1.

PG&E licensed operators continue the normal process of verifying that rod motion is proper for required movement using current plant-approved procedures. Also, an Operations Shift Order, dated June 22, 1993, discussed the Salem event and stated: "As always, the CO [control operator] should verify the expected response of the Rod Control System whenever moving rods."

PG&E Response to Item 2.

The rod deviation alarm was recently confirmed to be functional by PG&E using current plant-approved procedures.

PG&E Response to Item 3.

For enhanced operator awareness, an Operations Shift Order, dated June 22, 1993, was issued to all plant control operators advising them of the Salem event and the possibility for a similar event at Diablo Canyon Units 1 and 2.

PG&E Response to Item 4.

In response to the Westinghouse Owners Group survey, PG&E provided general rod control information and specific rod control operational experience. As part of the survey response, PG&E provided a copy of Licensee Event Report 2-86-007-00, "Reactor Trip Due to NIS Negative Rate Trip Signal Caused by Two Dropped Rods During the Investigation of a Rod Control Urgent Failure Alarm," (PG&E Letter No. DCL-86-090, dated April 4, 1986) because of its apparent similarity to the Salem event. It should be noted that the PG&E event involved outward movement of a complete rod bank rather than unexpected outward movement of individual rods of a bank as identified at Salem.

The PG&E event took place on March 5, 1986, with the Unit in Mode 1 (Power Operation) and involved an automatic reactor trip, with a subsequent turbine trip. While troubleshooting a rod control urgent failure alarm, a manual one-step insertion on Control Bank D was attempted. Bank demand counters for Bank D indicated an outward step. The manual rod insertion was attempted again and the bank demand counters indicated another step out. For both insertion attempts, the digital rod position indication was within its deadband. Upon again attempting to drive rods inward, operators noted two dropped Bank D rods. The dropped rods caused a nuclear instrumentation system negative rate reactor trip.

The LER describes the root cause of the dropped rods as a faulty logic module in the rod control circuity. On March 5, 1986, a fuse and the faulty module were replaced. On March 6, 1986, Control Bank D rods were successfully moved in and out under manual control. The rod control system was returned to normal and declared operable.