ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8909260013 DOC.DATE: 89/09/19 NOTARIZED: YES DOCKET #
FACIL:50-275 Diablo Canyon Nuclear Power Plant, Unit 1, Pacific Ga 05000275
50-323 Diablo Canyon Nuclear Power Plant, Unit 2, Pacific Ga 05000323
AUTH.NAME AUTHOR AFFILIATION
SHIFFER,J.D. Pacific Gas & Electric Co.
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

See July for WCAP 12221 & WCAP 12222

SUBJECT: Forwards application for amends to Licenses DPR-80 & DPR-82. Proprietary WCAP-12221 & non-proprietary WCAP-12222 encl.

DISTRIBUTION CODE: AP01D COPIES RECEIVED:LTR 1 ENCL 3 SIZE: 2+96
TITLE: Proprietary Review Distribution - Operating Reactor

NOTES: Withholding Granted 8/20/90

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD5 LA	1	1	PD5 PD	1	1
	ROOD,H	3	3			
INTERNAL:	ACRS	6	6	AEOD/DOA	1	1
	AEOD/DSP/TPAB	1	1	OGC/HDS2	1	0
	REG FILE 01	1	1			
EXTERNAL:	LPDR	1	1 NP	NRC PDR	1	1 NP

NSIC

A+A-2

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR ~~17~~ 18 ENCL ~~14~~ 17

R I D S / A D D S

R I D S / A D D S

Ca

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 19, 1989

PG&E Letter No. DCL-89-244

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 89-12
Revision of Technical Specifications to Delete the Steam Generator Water Level-Low Coincident with Steam/Feedwater Flow Mismatch Trip

Gentlemen:

Enclosure 1 is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82. The enclosed License Amendment Request (LAR) proposes to delete the Technical Specifications requiring a reactor trip in the event of steam generator water level-low coincident with steam/feedwater flow mismatch (low feedwater flow reactor trip) following installation of a new digital feedwater control system. Specific revisions are proposed to Technical Specification Table 2.2-1 and associated Bases and Tables 3.3-1, 3.3-2, and 4.3-1.

A similar license amendment was issued for Prairie Island Units 1 and 2 on April 3, 1989 (Amendment 87 to DPR-42 and Amendment 80 to DPR-60).

PG&E plans to install a new digital feedwater control system in Diablo Canyon Units 1 and 2 during the next refueling outages (October 1989 for Unit 1 and February 1990 for Unit 2). The new feedwater control system includes a steam generator level median signal selector (MSS). The MSS selects the median of three steam generator narrow range level signals and improves feedwater control performance and reliability. The MSS prevents interaction between the feedwater control and reactor protection systems by providing functional isolation between the two systems in accordance with the requirements of IEEE 279-1971.

PG&E believes the priority for NRC Staff review and approval of this LAR is high because elimination of the low feedwater flow trip and implementation of the MSS feature will (1) minimize potential reactor trips due to the low feedwater flow trip and (2) reduce surveillance test requirements and potential challenges to the reactor protection systems. Deletion of the low feedwater flow reactor trip will be made as soon as practical following installation of the new digital feedwater control system and issuance of a license amendment by the NRC.

CHANGE	LTR	ENCL
LPDR	1	1 NP
NSIC	1	1 NP
NRC PDR	1	1 NP

APO 1
3 PROP
1 3N-PROP

Attachment A

REVISION OF TECHNICAL SPECIFICATIONS TO DELETE THE STEAM GENERATOR WATER LEVEL-LOW COINCIDENT WITH STEAM/FEEDWATER FLOW MISMATCH TRIP

A. DESCRIPTION OF AMENDMENT REQUEST

This license amendment request (LAR) proposes to delete the requirement for the steam generator water level-low coincident with steam/feedwater flow mismatch reactor trip (low feedwater flow reactor trip), following installation of a new digital feedwater control system, from the following Technical Specifications (TS):

1. TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," Functional Unit 14, and associated Bases.

2. TS Table 3.3-1, "Reactor Trip System Instrumentation," Functional Unit 14.

3. TS Table 3.3-2, "Reactor Trip System Instrumentation Response Times," Functional Unit 14.

4. TS Table 4.3-1, "Reactor Trip System Instrumentation Surveillance Requirements," Functional Unit 14.

Changes to the Technical Specifications are noted in the marked-up copy of the applicable specifications (Attachment B).

B. BACKGROUND

The current Diablo Canyon reactor trip functions associated with the steam generator protection system are the steam generator water level low-low reactor trip and the low feedwater flow reactor trip.

The water level low-low reactor trip uses three level detectors in the protective channels for each of the four steam generators. If two of three channels on any single generator sense a low-low water level condition, a reactor trip will occur. This 2 of 3 coincidence trip logic on 1 of 4 steam generators cannot be blocked. The 2 of 3 coincident steam generator water level low-low signal will also provide the starting signal for the motor-driven and the turbine-driven auxiliary feedwater pumps. The basic function of the water level low-low trip channels is to preserve the steam generator as a heat sink for removal of residual heat. This automatic protection action is taken to ensure sufficient inventory in the steam generators to maintain the heat sink, reduce the capacity and starting time requirements of the auxiliary feedwater system, and to minimize the thermal transient on the reactor coolant system (RCS).

The low feedwater flow reactor trip signal is developed from three different parameters associated with each steam generator (steam flow, feedwater flow, and level). The logic circuitry is such that a trip will occur at the low level setpoint if feedwater flow in that steam generator is below the steam flow by a prescribed amount. Two of three narrow-range channels are used in the low water level circuitry with the other channel used for steam generator water level control. If one of two channels indicates less than the low water level setpoint, a low level signal exists from that steam generator. FSAR Update Figure 7.2-1 (Sheet 7 of 16) shows the reactor trip logic. The two steam flow signals, originating at the steam generator outlet nozzle, are sent to a mismatch circuit. The two feed flow signals originating at the inlet of each steam generator are sent to the same mismatch circuit. One steam flow is compared to its selected feed flow and the other to the remaining feed flow signal. Thus, the mismatch circuit for each steam generator continuously makes two separate comparisons. If either comparison indicates that feedwater flow has dropped 1.45×10^6 lbm/hr below steam flow, a feedwater flow/steam flow mismatch signal exists for that steam generator. The combination of a low steam generator water level signal and feedwater flow/steam flow mismatch signal from any one of the four steam generators causes a reactor trip. The low feedwater flow trip cannot be bypassed. This trip anticipates a loss of reactor heat sink.

The low feedwater flow reactor trip is used in conjunction with the steam generator water level low-low reactor trip to satisfy 10 CFR 50.55a, Codes and Standards, Paragraph (h), Protection Systems. This regulation endorses IEEE Standard 279-1971 (edition applicable to Diablo Canyon), "Criteria for Protection Systems for Nuclear Power Generating Stations." Section 4.7.3 of IEEE 279 requires that where a single random failure can cause a control system action that results in a generating station condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining redundant protection channels shall be capable of providing the protective action even when degraded by a second random failure. The low feedwater flow reactor trip function satisfies this interaction criterion. The low feedwater flow reactor trip is not required to mitigate the consequences of any accident described in the FSAR Update accident analyses or in the response to any NRC Bulletins or other correspondence to the NRC.

During the next refueling outages for Units 1 and 2, PG&E plans to install a new digital feedwater control system as allowed by 10 CFR 50.59. The new system will include a steam generator level median signal selector (MSS) that selects the median of three steam generator narrow range signals. This improved design will enhance feedwater control performance and reliability. The installation of the MSS effectively eliminates the concern regarding a single random failure causing a control system action that results in a condition requiring protective action and preventing proper operation of a protection system demand designed to protect against this condition. Thus, MSS will prevent interaction between the feedwater control and reactor protection

systems in accordance with the requirements of IEEE 279-1971. Removal of this interaction nullifies the need for the low feedwater flow reactor trip. Thus, the MSS will functionally separate steam generator narrow range level protection channels (low-low trip) to provide compliance with IEEE 279-1971 and satisfy the original design basis. Additional details of the MSS design and evaluation are provided in WCAP-12221.

Electrical separation of the control and protection circuits is through Westinghouse Model 131-110 isolation amplifiers, which are already qualified and used extensively in safety-related isolation applications at Diablo Canyon. These isolators were functionally tested by Westinghouse in April 1970 and documented in WCAP-7509-L. Three standard producton isolation amplifiers, Model 131-110, manufactured by Westinghouse Hagan Computer Systems Division, were tested at normal ambient temperature for isolation characteristics. The isolation amplifiers maintained input circuit integrity (protection circuit) for all disturbances applied both common mode and normal mode to the output circuit. In some cases, the output stage was damaged by the applied disturbance, but the protection circuit remained functional and within design accuracy limits.

Seismic qualification of the Westinghouse Model 131-110 was performed in 1984 to verify the ability to meet Hosgri, modified Double Design and Design Earthquake response spectra. Westinghouse performed the test by qualifying the Hagen Racks as a single unit and monitoring equipment operation. A letter documenting seismic qualification of the 7100 Process Control and Protection Racks was transmitted to PG&E on November 1, 1984.

The MSS uses self checking of inputs and outputs to verify proper control system inputs. During signal selector testing, input signal out-of-range checks will be performed for each narrow range steam generator level signal. This consists of ensuring that "BAD" data status is assigned to the input if its value is out of the normal range (1-5V). The bi-stable values for BAD data are less than 0.9 volts and greater than 5.1 volts. Furthermore, testing will be performed to verify that with two or more level channels BAD (i.e., out-of-range), the control system switches to manual.

With respect to the potential of a protection system failure propagating to the control system such that the function is negated, a number of system design measures have been implemented to protect against this occurrence.

- The three narrow range level channels are physically distributed (separated) among three different input cards. Thus, a protection system failure may affect only one of three level input channels. The MSS which resides in the software will continue to select the proper value.

- The three different input cards are physically distributed (separated) among three different input card crates. A single card being physically damaged cannot subsequently damage another level channel card.

- The input cards are also isolated through a transformer between the analog inputs and the digital portion of the system such that an out-of-range signal that may damage the input card will not propagate through to the central processing unit.

- Multiple narrow range level channel failures (either by sensor or input card failure) will result in switch-to-manual control, effectively negating the control/ protection interaction issue.

The above "defense in depth" design items effectively prevent a single protection fault from disabling the MSS. PG&E concludes that no credible protection system failure can result in a failure of the MSS or an upset of the feedwater control system.

C. JUSTIFICATION

As discussed above, the MSS combined with the steam generator water level low-low reactor trip satisfies the requirements of Section 4.7.3 of IEEE 279-1971, thereby eliminating the need for the low feedwater flow reactor trip. As noted, the low feedwater flow reactor trip is not required to mitigate the consequence of any accident described in the FSAR Update accident analyses or any correspondence to the NRC.

Elimination of the low feedwater flow reactor trip will reduce periodic surveillance test requirements and potentially reduce the number of unnecessary challenges to the reactor protection system.

D. SAFETY EVALUATION

As described above, the low feedwater flow reactor trip is not required to mitigate the consequences of any analyzed accidents in the FSAR Update or in the response to any NRC Bulletins or other correspondence to the NRC. Installation of the MSS in the digital feedwater control system will allow the control and protection interaction requirement of IEEE 279-1971 to be met without reliance on the low feedwater flow reactor trip. The steam generator water level low-low trip will provide the necessary protection when required. The low-low trip combined with the MSS eliminates the need for the low feedwater flow reactor trip. Removal of the low feedwater flow trip will reduce the probability of unplanned reactor trips and unnecessary challenges to the safeguard systems.

Therefore, PG&E believes that there is reasonable assurance that the health and safety of the public will not be adversely affected by elimination of the low feedwater flow reactor trip.

E. NO SIGNIFICANT HAZARDS

PG&E has evaluated the no significant 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 a testing facility involves no significant hazards considerations, 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 no significant hazards consideration standards.

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

   Deletion of the low feedwater flow reactor trip combined with installation of the MSS in the digital feedwater control system will allow continued compliance with the requirement of IEEE 279-1971 for prevention of control and protection system interaction. The low feedwater flow reactor trip is not required to mitigate the consequences of any analyzed accidents in the FSAR Update accident analyses. Removal of the low feedwater flow trip will reduce the probability of unplanned reactor trips and unnecessary challenges of engineered safety feature systems.

   Therefore, the proposed changes do 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?

   The changes involve removal of the low feedwater flow reactor trip function and the addition of the MSS function. With the MSS installed, the steam generator water level low-low reactor trip will provide protection in compliance with IEEE 279-1971. As discussed above, the low feedwater flow reactor trip function is not required to mitigate the consequences of any accident described in the FSAR Update accident analyses.

Abnormal operational transients that could result from feedwater control system failures are analyzed in the FSAR Update. These analyses bound all postulated feedwater control system failures.

The new digital feedwater control system will improve feedwater performance and reliability. Evaluation of the new system in accordance with 10 CFR 50.59 indicates that no credible protection system failure will result in failure of the MSS or an upset of the feedwater control system.

Therefore, the proposed changes do 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?

As discussed above, the low feedwater flow reactor trip is not assumed in the FSAR Update accident analyses. A new digital feedwater control system and MSS function will significantly reduce the potential for feedwater control system related reactor trips and unnecessary challenges to the reactor protection system.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, PG&E concludes 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 offsite, 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.