NOV 15 1983

Docket Nos.: 50-275
and 50-323

Mr. J. O. Schuyler, Vice President
Nuclear Power Generation
c/o Nuclear Power Generation, Licensing
Pacific Gas and Electric Company
77 Beale Street, Room 1435
San Francisco, California 94106

Dear Mr. Schuyler:

Subject: NRC Implementation Dates for Amendment No. 1 to Facility Operating License DPR-76 (Diablo Canyon Nuclear Power Plant, Unit 1)

The Nuclear Regulatory Commission issued Amendment No. 1 to Facility Operating License No. DPR-76 for the Diablo Canyon Nuclear Power Plant, Unit 1 on February 5, 1982. Amendment No. 1 stated that implementation dates for the four license conditions delineated below would be established by the NRC prior to loading fuel.

No. 1 - 2.C.(8)h — Post Accident Sampling (Section II.B.3)

PG&E shall complete corrective actions needed to provide the capability to promptly obtain and perform radioisotopic and chemical analyses of reactor coolant and containment atmosphere samples under degraded core conditions without excessive exposure. An implementation date for this item will be established by the NRC prior to fuel loading in Unit No. 1.

No. 2 - 2.C(8)k — Additional Accident Monitoring Instrumentation (Section II.F.1)

PG&E shall install continuous indication in the control room of the following parameters:

(1) Containment radiation monitors.

(2) Noble gas effluent from each potential release point.

An implementation date for this item will be established by the NRC prior to fuel loading in Unit No. 1.

No. 3 - 2.C(8)l Instruments for Inadequate Core Cooling (Section II.F.2)

PG&E shall resolve the issue on plant computer isolation devices. PG&E shall upgrade the incore thermocouple system, except for the incore thermocouple in-containment connectors and junction boxes. PG&E shall replace the incore thermocouple in-containment connectors and junction boxes during the first extended outage following component availability. An implementation date for this item will be established by the NRC prior to fuel loading in Unit No. 1.

No. 4 - 2.C(8)o Calculations for Small-Break LOCAs (Section II.K.3.30 and II.K.3.31)

PG&E is participating in the Westinghouse Owner's Group effort for this item and shall conform to the results of this effort. The analysis for model justification shall be submitted. An implementation date for this item will be established by the NRC prior to fuel loading in Unit No. 1.

In regard to Item No. 1, by letters dated August 31 and November 11, 1983 the Pacific Gas & Electric Company (PG&E) stated that the use of the interim post-LOCA sampling system together with the containment air sampling component of the permanent post-LOCA sampling system satisfies NUREG-0737 requirement. Testing of the liquid sampling system will be accomplished when the reactor coolant system is pressurized. Procedures have been provided and individuals have been trained in the operation of these sampling systems. The licensee also stated that its interim procedures for estimating core damage will be revised, prior to exceeding five percent of rated power, to address (1) normal operation, (2) macroscopic fuel clad damage, (3) severe fuel overtemperature, and (4) fuel melting. The staff finds the combined use of the interim and permanent post-LOCA sampling systems acceptable. Because of the low fission product inventory resulting from low power operation, not to exceed five percent of rated power, the current interim procedures for estimating core damage are acceptable, however we will require that interim procedures be revised to address the above four features prior to exceeding five percent of rated power. The staff finds that the actions taken satisfy the license conditions for post accident sampling.

In regard to Item No. 2, the licensee has stated in its November 11, 1983 letter that the containment radiation monitors and the mid-range and high-range noble gas effluent monitors for the plant vent and the main steamline safety relief/dump valves have been installed and are operational. Furthermore, continuous readout of both the containment radiation monitors and the noble gas effluent monitors is provided in the control room. The licensee stated that the steam generator blowdown tank vent is used only intermittently and, if in use, would be isolated by any one of three plant radiation monitors upon alarm and obviates the need for installation of a mid-high range noble gas monitor in the blowdown tank vent. The staff finds that because the blowdown tank is used intermittently and will be isolated upon the alarming of any one of three plant radiation monitors that this is not a credible release path following a loss of coolant accident and that the installation of a noble gas effluent monitor on this vent is not required. The staff finds, since the licensee has fully completed the requirements of this condition in a satisfactory manner, the license condition for Item No. 2 above has been met.

With respect to Item No. 3, PG&E stated in its August 31 and November 11, 1983 letters that all work pertaining to the plant computer isolation devices has been completed. PG&E also stated in the above cited letters that the installation of the upgraded incore thermocouple system has been completed, and that testing and calibration will be completed during low power testing. This satisfies the license condition requiring upgrading of the incore thermocouple system. Moreover, by letter dated September 16, 1983, PG&E stated that the incore thermocouple in-containment connectors and junction boxes have been replaced. Therefore, all the requirements of license condition 2.C(8)1 (see SSER No. 14, Section II.F.2) have been satisfied and no further action is required.

For Item No. 4 above, the NRC staff is treating this on a generic basis. The staff is currently reviewing the Westinghouse Corporation generic submittal (NUREG-II.K.3.30). We require that the Pacific Gas & Electric Company submit its plant specific analysis (NUREG-II.K.3.31) which must be approved by the NRC staff within one year from the date of NRC approval of the Westinghouse generic models. A satisfactory submittal could reference applicable calculations in II.K.3.30.

Sincerely,

"Original Signed By:

"

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

cc: See next page

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For Item No. 4 above, the NRC staff is treating this on a generic basis. The staff is currently reviewing the Westinghouse Corporation generic submittal (NUREG-II.3.K.30). We require that the Pacific Gas & Electric Company submit its plant specific analysis (NUREG-II.K.3.31) which must be approved by the NRC staff within one year from the date of NRC approval of the Westinghouse generic models.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

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see note

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