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AUTH.NAME AUTHOR AFFILIATION
SHIFFER,J.D. Pacific Gas & Electric Co.
RECIP.NAME RECIPIENT AFFILIATION
KNIGHTON,G.W. Licensing Branch 3

SUBJECT: Forwards response to 850314 request for addl info re proposed Tech Specs.Ack of receipt of matls requested.

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PG&E 77 BEALE STREET • SAN FRANCISCO, CALIFORNIA 94106 • (415) 781-4211 • TWX 910-372-6587

JAMES D. SHIFFER
VICE PRESIDENT
NUCLEAR POWER GENERATION

April 11, 1985

PGandE Letter No.: DCL-85-147

Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Docket No. 50-323
Diablo Canyon Unit 2
Proposed Unit 2 Technical Specifications - Additional Information

Dear Mr. Knighton:

On March 14, 1985, the NRC Staff requested PGandE to respond to questions arising from their review of the proposed DCPP Unit 2 Technical Specifications. The enclosure provides the requested responses.

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

Sincerely,

JDShiffer

Enclosure

cc: R. T. Dodds
J. B. Martin
H. E. Schierling
Service List

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1/1

ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION REGARDING
DIABLO CANYON UNIT 2 TECHNICAL SPECIFICATIONS

In a letter dated March 14, 1985, the NRC Staff requested PGandE to respond to questions regarding the proposed Diablo Canyon Unit 2 Technical Specifications. The five Staff questions are listed below; each question is followed by PGandE's response.

NRC QUESTION 1

Table 3.3-2, Reactor Trip System Instrumentation Response Times (page 3/4 3-8)

The overpower delta T trip is relied upon to mitigate main steam line breaks in the FSAR. For Item 8 of Table 3.3-2, a N.A. is provided indicating that the trip is not relied upon. Justify this apparent inconsistency.

PGandE RESPONSE

FSAR subsection 15.4.2.1.1, page 15.4-7 presents a listing of available protection items, including the overpower delta T trip. However, a review of the Diablo Canyon main steam line break analysis by Westinghouse has indicated that the overpower delta T trip is not used in the analysis. Technical Specification Table 3.3.2 is, therefore, consistent with the FSAR analysis.

NRC QUESTION 2

Section 4.4.1.2.2 and 4.4.1.3.2, Operability of the Steam Generators (pages 3/4 4-2 and 3/4 4-4)

These LCOs and surveillance requirements state that the required steam generator(s) should be determined operable by verifying the secondary side water level to be 15% at least once per 12 hours. Provide the basis for the 15% steam generator water level and specify whether the 15% level is measured on the narrow range or the wide range instrumentation.

PGandE RESPONSE

The 15% steam generator water level is used as a reference in determining the operability of the steam generator(s) to ensure that sufficient decay heat

removal capability is available when using the secondary system for heat removal.

The 15% narrow range level value was selected as the reference value to reflect operability of the steam generators because it corresponds to the low-low steam generator water level reactor trip setpoint. The basis for this trip states, "The Steam Generator Water Level Low-Low Trip Protects the Reactor from Loss of Heat Sink..."

Auxiliary feedwater is normally supplied to the steam generators in Modes 3 and 4 until the residual heat removal system is placed in service. In Mode 3, if auxiliary feedwater is not in service, upon reaching the 15% level setpoint, the auxiliary feedwater will be automatically initiated (Ref. proposed Technical Specification 3/4.3.2, Table 3.3-3, Item 6).

Additionally, the 15% value is below the normal zero power programmed steam generator water level (33%), yet it is sufficiently high to ensure that: (1) the steam generator tubes are covered, and (2) a sufficient volume of water exists in the steam generator to allow for decay heat removal capability when using the secondary system for heat removal.

It should be noted that, at the 15% narrow range level, the top of the steam generator tubes are covered by more than 4 feet of water. Removal of decay heat in Mode 4 requires that only a small length of the tube bundle be covered with water. Therefore, substantial margin is provided by the 15% narrow range level value.

In conclusion, sufficient basis exists for the selection of a 15% narrow range steam generator level to ensure the operability of the steam generators in support of Limiting Conditions for Operation 3.4.1.2 and 3.4.1.3.

NRC QUESTION 3

Table 3.4.-1, Reactor Coolant System Pressure Isolation Valves (page 3/4 4-21)

Charging system discharge check valves are not included in the table of isolation valves for which leak surveillance will be performed ... include these valves or provide an analysis of loss of charging pump flow with the check valves stuck in the open position.

PGandE RESPONSE

The charging system discharge check valves should not be included in Table 3.4-1 as they do not present a realistic potential for contributing to an intersystem LOCA condition since the charging system is composed of reactor coolant system (RCS) pressure rated piping (with the exception of the pump suction piping). In fact, one of the functions of the charging system is to provide a pressure testing service to the RCS (Ref. Westinghouse SD-PGE/PEG-200/A, "Chemical and Volume Control System - System Description").

Table 3.4.-1 in the Technical Specifications, with its associated surveillance requirements, "...Provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent Intersystem LOCA." (Ref. Technical Specification Basis 3/4.4.6.2, "Operational Leakage.")

For a potential intersystem LOCA condition to exist in the charging system high pressure reactor coolant would need to communicate with lower pressure centrifugal charging pump suction piping. This would be impossible during normal operating conditions since the charging system would be operating with flow to the RCS. Therefore, for this situation to occur, assuming normal charging lineup and not taking credit for isolation valves, (1) the charging system would need to be secured (i.e., no charging flow, which is a very unusual condition), (2) reactor coolant would need to leak past closed check valves 8378B, 8379B, 8378C, and 8478A or 8478B, and through the idle pump, and (3) the suction line relief valve 8125 would need to fail to operate.

It should also be noted that the centrifugal charging pump discharge check

valves 8478A and 8478B are included in the IST valve program and checked quarterly to ensure they seat (closed) to prevent reverse flow through the idle centrifugal charging pump.

Furthermore, since the charging system will normally be in operation throughout the applicable Modes 1, 2, 3, and 4, and since Surveillance Requirement 4.4.6.2.2.c requires the verification of leakage to be within its limit,* the charging system would be placed in the position that the check valves would be required to be tested in the closed position and shown to be within leakage limits every 24 hours.

NRC QUESTION 4

Section 3.4.9.3, Overpressure Protection System (page 3/4 4-35)

In order to comply with Appendix G limits throughout the life of the plant, the PORV setpoint must be updated periodically to account for the irradiation of the pressure vessel. To ensure that the above setpoint curve is updated, propose a requirement for Surveillance 4.4.9.1.2 on page 3/4 4-30.

PGandE RESPONSE

The words "and the setpoint of Technical Specification 3.4.9.3.b" will be added to the end of the final sentence of Surveillance Requirement 4.4.9.1.2.

NRC QUESTION 5

Provide proposed technical specifications to correct the lack of technical specifications preventing a control rod withdrawal event in MODE 4 when the presently proposed technical specifications only require one RHR pump to provide coolant flow. In your response you should propose a technical specification requirement to remove power from the rod devices in this mode, or propose a requirement for a number of operating Reactor Coolant Pumps based on an approved safety analysis.

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* "...within 24 hours following valve actuation due to automatic or manual action or flow through the valve. After each disturbance of the valve, in lieu of measuring leak rate, leaktight integrity may be verified by absence of pressure buildup in the test line downstream of the valve."

PGandE RESPONSE

Both the DCPP Unit 1 Technical Specifications and the Proposed Units 1 and 2 Common Technical Specifications are consistent in this area with the guidance provided in NUREG-0452, Rev. 4 (and Draft Rev. 5), "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," which is described in the Standard Review Plan Chapter 16 as the document containing regulatory guidance that satisfies the NRC acceptance criteria.

PGandE believes that both DCPP Units 1 and 2 can be operated safely because the described control rod withdrawal event in Mode 4 will not occur. This assurance is provided by the following factors:

(1) The control rod drive mechanisms are not normally energized when reactor coolant pumps (RCPs) are not operating. Certain tests of the control rods are conducted without operation of the RCPs, but these tests are of limited duration and are closely controlled by approved procedures. In all other circumstances, plant adminstrative controls will assure that the control rod drives are not energized.

(2) Uncontrolled rod withdrawal during the test periods is highly unlikely. The rod control system permits automatic rod movement only in Mode 1 above 15% power. Manual operation of the control rods during test requires direct operator involvement in the manipulation of the controls. Adequate indications are available in the control room for the operator to promptly detect an uncontrolled rod withdrawal event and trip the reactor.

(3) During the periods when the control rod drives are energized for testing without two RCPs operating, the reactor coolant will be borated to 2000 ppm or greater. WCAP-10593, "The Nuclear Design and Core Physics Characteristics of the Diablo Canyon Unit 2 Nuclear Power Plant Cycle 1," indicates that reactor criticality cannot occur with all the control rods withdrawn when the boron concentration is at that level. In Mode 4, the

reactor must be maintained with a shutdown margin $\geq 1.6\% \Delta K/K$ (Technical Specification 3.1.1.1) and K_{eff} must be less than 0.99 (Definition, Table 1.1).

In conclusion, this concern has been adequately addressed by current plant practice, and additional Technical Specification requirements are unnecessary and unjustified.