March 30, 1998

Mr. Gregory M. Rueger, Senior Vice President
and General Manager
Pacific Gas and Electric Company
Nuclear Power Generation N9B
P.O. Box 770000
San Francisco, California 94177

SUBJECT: DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2 - TECHNICAL SPECIFICATION BASES CHANGE (TAC NOS. M99934 AND M99935)

Dear Mr. Rueger:

The staff has incorporated the revision of the Bases for Technical Specifications 3/4.6.2.3 "Containment Cooling System" and 3/4.7.1.2 "Auxiliary Feedwater System" as provided by your letter dated October 24, 1997, into the Diablo Canyon Technical Specifications. The revisions support extended fuel cycles to 24 months. The overleaf pages are provided to maintain document completeness.

Sincerely,

Original Signed By

Steven D. Bloom, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-275
and 50-323

Enclosure: Bases Pages

cc w/encl: See next page

DISTRIBUTION:
Docket
PUBLIC
PDIV-2 Reading
EAdensam (EGA1)
OGC
WBateman
EPeyton
JKilcrease, Region IV
LHurley, Region IV
JBianchi, WCFO (2)
GHill (4)
ACRS
KPerkins, WCFO
HWong, WCFO
JWermiel
WBeckner

Document Name: BASES.DC

OFC	PDIV-2	PDIV-2	HICB
NAME	EPeyton	SBloom	JWermiel
DATE	11/19/97	~~11/ /97~~ 3/30/98	03/30/98

OFFICIAL RECORD COPY

9804080324 980330
PDR ADOCK 05000275
P PDR

cc w/encl:
NRC Resident Inspector
Diablo Canyon Nuclear Power Plant
c/o U.S. Nuclear Regulatory Commission
P. O. Box 369
Avila Beach, California 93424

Dr. Richard Ferguson, Energy Chair
Sierra Club California
1100 11th Street, Suite 311
Sacramento, California 95814

Ms. Nancy Culver
San Luis Obispo
Mothers for Peace
P. O. Box 164
Pismo Beach, California 93448

Chairman
San Luis Obispo County Board of
Supervisors
Room 370
County Government Center
San Luis Obispo, California 93408

Mr. Truman Burns
Mr. Robert Kinosian
California Public Utilities Commission
505 Van Ness, Room 4102
San Francisco, California 94102

Mr. Steve Hsu
Radiologic Health Branch
State Department of Health Services
Post Office Box 942732
Sacramento, California 94232

Diablo Canyon Independent Safety
Committee
ATTN: Robert R. Wellington, Esq.
Legal Counsel
857 Cass Street, Suite D
Monterey, California 93940

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Harris Tower & Pavillion
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

Christopher J. Warner, Esq.
Pacific Gas & Electric Company
Post Office Box 7442
San Francisco, California 94120

Mr. Robert P. Powers
Vice President and Plant Manager
Diablo Canyon Nuclear Power Plant
P. O. Box 56
Avila Beach, California 93424

Telegram-Tribune
ATTN: Managing Editor
1321 Johnson Avenue
P.O. Box 112
San Luis Obispo, California 93406

3/4.6.2.3 CONTAINMENT COOLING SYSTEM (Continued)

In order for the two CFCUs to remove sufficient heat to perform their intended function, 2000 gpm CCW flow must be supplied to the CFCU cooling coils. Analysis has determined that if 1600 gpm flow is supplied to the CFCU cooling coils during normal operation with the nonvital CCW header in service, at least 2000 gpm will be supplied to the CFCU cooling coils during LOCA coincident with a failure of vital Bus G.

The CCW system configuration during normal operation is different from the configuration during emergency core cooling system actuation. Nonvital header C is automatically isolated in most accident scenarios. This results in increased flow to the remaining components supplied by the two vital headers. Cooling water flow to the CFCU of 1650 gpm established in the normal plant configuration with non vital header C in service and the RHR heat exchangers isolated will result in CCW flow greater than or equal to 2000 gpm during accident conditions coincident with a Bus G failure.

One postulated single failure, the failure of vital Bus H, will prevent automatic isolation of nonvital header C because the power supply for the isolation valve is provided from Bus H. Nonvital header C being open is a different condition from that for the license basis containment pressure analysis described in supplemental safety evaluation report (SSER) 16 and FSAR, Section 6.2B.3, page 6.2B-5. For this accident scenario, the CCW flow to the CFCU coils following the accident will not change significantly from the observed flow during normal operation.

The effects of this case on containment integrity have been analyzed. The H Bus failure consequences, using mechanistic assumptions (ie., the components on other powered buses are assumed to operate and the components on Bus H have no power and are assumed to not operate) and a single failure, show that a CFCU cooling flow rate of 1650 gpm is adequate to perform the CFCU heat removal function for this scenario. The consequences of this scenario remain bounded by the license basis analysis.

If a single failure of Bus F is assumed, nonvital CCW header C will isolate and at least 2000 gpm CCW flow will be supplied to the CFCUs.

A footnote to the surveillance requirement specifies that operation of the CFCUs is permitted with low component cooling water (CCW) flow to the CFCUs due to ASME Section XI testing required by TS 4.0.5 or decay heat removal in Mode 4 with the residual heat removal heat exchangers in service. To support this conclusion, a calculation was performed. This calculation evaluated containment heat removal with one train of containment spray OPERABLE and reduced CCW flow to three CFCUs. The calculation concluded that this configuration would provide adequate heat removal to ensure that the maximum design pressure of containment was not exceeded during a DBA in Mode 1. This analysis also determined that a single failure could not be tolerated during this condition and still assure that the maximum design pressure of containment would not be

3/4.6.2.3 CONTAINMENT COOLING SYSTEM (Continued)

exceeded. Since a single failure cannot be tolerated, the footnote limits the acceptability of low CCW flow to the CFCU cooling coils to Mode 4 with the RHR system in service and ASME Section XI testing in Modes 1 through 4.

In order to support the analysis that permits operation with low CCW flow to the CFCUs, both containment spray trains must be OPERABLE and at least three CFCU must be verified OPERABLE prior to opening an RHR heat exchanger outlet valve for Section XI testing.

Surveillance Requirement 4.6.2.3a.3)

TS 4.6.2.3a.3) requires that each CFCU be started in low speed every 31 days. The purpose of this requirement is to assure that the CFCU and the associated control equipment is capable of operating in the configuration required for the DBA. The surveillance frequency of 31 days is based on the known reliability of the fan units and controls, redundancy available, and the low probability of significant degradation of the CFCUs occurring between surveillances.

Surveillance Requirement 4.6.2.3b.

TS 4.6.2.3b. requires that each CFCU be started on a safety injection signal once per REFUELING INTERVAL. This surveillance provides assurance that the circuitry required to start the CFCU during a DBA is OPERABLE. The REFUELING INTERVAL frequency is based on the need to perform these surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillances when performed at the REFUELING INTERVAL frequency. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 40, GDC 41, GDC 42, and GDC 43.
2. 10 CFR 50, Appendix K.
3. FSAR Section 6.2B.3
4. FSAR Section 6.2.1.3.6
5. FSAR Table 6.2-5
6. FSAR Section 6.2.2.2.2.2
7. FSAR Section 9.2.2
8. FSAR Section 15.4

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM (Continued)

Action b.

If two AFW trains are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within the following 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

Action c.

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

SURVEILLANCE REQUIREMENTS

Surveillance Requirement 4.7.1.2.1a.

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

Surveillance Requirement 4.7.1.2.1b.

Testing each AFW pump pursuant to TS 4.0.5 ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM (Continued)

of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 2) (only required at 3 month intervals) satisfies this requirement. The 31 day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.

This SR is modified by a Note indicating that the SR for the turbine driven pump should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

Surveillance Requirement 4.7.1.2.1c.

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The REFUELING INTERVAL Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The REFUELING INTERVAL Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The REFUELING INTERVAL Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by a Note indicating that the SR for the turbine driven pump should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

REFERENCES

1. FSAR, Section 6.5 and Section 15.2.8

2. ASME, Boiler and Pressure Vessel Code, Section XI

3. Surveillance Test Procedures (STPs) P-AFW-11, P-AFW-12, P-AFW-13, P-AFW-21, P-AFW-22, P-AFW-23