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TO: Mr. John F. Stolz	FROM: Pacific Gas & Elec. Company, San Francisco, California, Philip A. Crane, Jr.	DOCKET NUMBER: 50-275/323
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		DATE OF DOCUMENT: 12/1/77
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DESCRIPTION

(1-P)

PLANT NAME: Diablo Canyon Units 1 & 2
RJL 12/6/77

ENCLOSURE

Consists of responses to two NRC questions on the Unit No. 1 structural integrity test......

(3-P)

40 ENCL *

FOR ACTION/INFORMATION

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December 1, 1977

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U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Mr. John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Docket No. 50-275-OL
Docket No. 50-323-OL
Diablo Canyon Units 1 & 2

Dear Mr. Stolz:

Attached to this letter are our responses to the two questions on the Unit 1 structural integrity test which were enclosed in your letter of August 24, 1977.

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

Very truly yours,

Philip A. Crane, Jr.

Enclosures
CC w/enc.: Service List

773400061

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON SITE, UNIT NO. 1
CONTAINMENT STRUCTURE STRUCTURAL INTEGRITY TEST

Additional Information Requested by
USNRC Letter Dated August 24, 1977

Item 10

Your response to Item 7 did not provide any factual data from actual structural integrity tests upon which the conclusion could be drawin that the 20% margin of error for calculated displacements is indeed acceptable as you concluded. Furthermore, since your response to Item 2 suggests that the establishing of the 20% margin of error was based on a study of some data it is necessary that this data be provided to the staff to justify your conclusion. While the 20% margin of error may prove to be quite adequate you must provide an adequate basis, preferably from the structural integrity test of other containments, upon which such a margin can be established.

Response

We considered several factors affecting prediction accuracy:

a) Residual stress due to concrete shrinkage

A minimum shrinkage strain $\mathcal{E}_s$ of 0.0002 inches per inch was assumed. The resulting compressive strain in reinforcing is relieved when concrete is cracked during the test, causing additional displacement $\triangle_s$.

$$\triangle_s = \mathcal{E}_s R = 0.0002 \times 876 = 0.175''$$

(R = 876" = radius of hoop reinforcing)

b) Measurement Errors

The measurement accuracy is 1/32". If errors of this magnitude are made at zero and 54 psi pressure, the total error is $\triangle_e = 1/16'' = 0.063''$

c) Temperature Varations

$10^{\circ}F$ temperature difference results in radial expansion of
$\triangle_t = (\mathcal{E}_t)\ (t)\ (R) = 5.5 \times 10^{-6} \times 10 \times 884 = 0.049''$

Where:

ε_t = concrete thermal expansion coefficient

t = temperature differential

R = Containment radius

d) Out of Roundness Adjustment

"As-built" deviations of the containment shell from a circular shape were a maximum of four inches. Relatively large shape adjustments were expected as the structure sought a circular configuration under pressure. We assumed the average of the six measurements at each level to represent radial displacement and deviations from the average to represent shape adjustments. But with only six measurements over a 460 foot circumference, inward and outward adjustments might not average out. Thus we felt it reasonable to allow 1/8" to account for this "out of roundness" adjustment.

The total of these factors is 0.412". This represents 44 percent of the maximum calculated radial displacement of the containment which seemed excessive. Therefore a study of other containment integrity tests was made to determine if more stringent limits might reasonably be established.

This study showed the 20% margin used on the Robert E. Ginna test was the most stringent limit successfully met. Considering that the most adverse simultaneous occurrence of all factors was unlikely it is reasonable to combine them on a square root-sum of squares basis. The maximum error in this case is 0.225" which represents a 24% margin. Taking this and the successful Ginna experience into account a 20% margin was adopted.

Item 11

In your response to Item 9 you made several statements on Page 5 in the paragraph entitled "Strains, Stresses" which need explanation:

(1) The purpose of the structural integrity test is to verify the behavior of the structure predicted by the analysis. Consequently, some acceptance standard should be set up prior to the structural integrity test being performed. Your statement that "no such standards were set for strain gage readings and no attempt was made to predict them". should be explained.

(2) You indicated that some readings were used for evaluation of the structural performance and others rejected. The report also contains a statement that the readings which have been used "appear consistent with other recorded data and credible..." This statement suggests that you are in possession of some additional information or "recorded data" which you used as an acceptance standard.

You are requested to provide the above noted specific information.

Response

(1) Our study of other tests showed only very general correlation between strain gage readings and the test variables. We observed large differences in readings of gages with theoretically identical stress and erratic stress variations with changing internal pressure. Large margins of error compromising the test objectives would be needed to allow for these discrepancies. For this reason, we concluded it was not practical to set meaningful acceptance criteria beforehand, without the benefit of a thorough evaluation and interpretation of the strain gage test data as well as consideration of the other data obtained during the test such as displacement and cracking patterns. We feel the basic objective of strain gage instrumentation - a complete evaluation of strain distribution - has been accomplished.

(2) The statement "appear consistent with other recorded data and credible..." refers to a good correlation between strains observed from two or more independent measurements and calculated values. A sample of such a correlation was given in the table on Page 7 of our response, where strain gage measurements at elevations 161 and 236 compare closely with the corresponding displacement measurements at the same elevations.

All such data is given in our test report, except that the strain gage data is given for the maximum pressure of 54 psi only. We made 23 separate strain gage data runs which provided a complete time history of readings at each gage. Malfunctioning gages could be readily identified by erratic changes of readings throughout the test and thus their credibility evaluated.

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Docket Nos. 50-275
and 50-323

Pacific Gas and Electric Company
ATTN: Mr. John C. Morrissey
Vice President & General Counsel
77 Beale Street
San Francisco, California 94106

bcc: ACRS (16)
NSIC
TIC

Gentlemen:

SUBJECT: PRESSURE VESSEL FRACTURE TOUGHNESS PROPERTIES (DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2)

Our review of data received from reactor vessel material surveillance programs indicates that the materials used in the fabrication of older pressure vessels may have a wider variation in sensitivity to radiation damage than originally anticipated. In addition, some reactor vessels incorporate more than one heat of materials, including weld metals, in their beltline regions, but all of these heats may not be included in the reactor vessel material surveillance program.

Although our review of these data does not reveal a basis for concern regarding reactor vessel integrity over several years of operation, the information does indicate a need for the detailed review of the materials employed in reactor vessel construction (in light of this recent data) and a review of the specimens employed in the surveillance program to determine if the present specimens reasonably represent the limiting materials in the reactor vessel beltline region.

The staff has determined that additional information is required on materials in the beltline region of the reactor vessel(s) at your facility in order for us to complete our evaluation of the potential for developing marginal fracture toughness properties after a period of reactor operation and to assess the need for augmented material surveillance programs.

Accordingly, you are requested to provide a response to the information listed in the enclosure within 60 days of receipt of this letter.

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SURNAME➤						
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[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

[illegible]

NOV 23 1977

This request for generic information was approved by GAO under a blanket clearance number R0071. This clearance expires September 30, 1978.

Sincerely,

Original Signed by
John F. Stolz
John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management

Enclosure: Questionnaire

cc w/enclosure:
See next page

OFFICE	LWR 1	LWR 1				
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DATE	11/ /77	11/ /77				

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ENCLOSURE

QUESTIONNAIRE FOR FRACTURE TOUGHNESS PROPERTIES OF
OLDER REACTOR VESSELS

121.0 MATERIALS ENGINEERING BRANCH - MATERIALS INTEGRITY SECTION

121.1 Provide the purchase order date for your reactor vessel, identify the firm or firms with whom the purchase order was placed, the vessel fabricator, and applicable edition of the ASME Code requirement pursuant to 10 CFR Part 50.55a(c).

121.2 Identify each material (plate, and/or forging and weld metal) in the beltline region (as defined by paragraph II.H, Appendix G, 10 CFR Part 50) and provide a sketch showing the location of these materials in the reactor vessel. Provide the following information for each material:

(1) Chemical analyses; particularly those elements known to affect irradiation sensitivity and degrade the upper shelf fracture energy (Cu, P, and S).

(2) Unirradiated fracture toughness properties (T_{NDT}, RT_{NDT} and upper shelf fracture energy) as required by Appendix G, 10 CFR Part 50, identifying the limiting material in the reactor vessel beltline region.

(3) Estimate the maximum anticipated change in RT_{NDT} and upper shelf fracture energy as a function of the EOL fluence at the inner wall for the materials in the beltline region of the reactor vessel.

121.3 Describe the surveillance program for the reactor vessel(s), list the materials (plate, and/or forging and weld metal) and justify their selection. State any deviation from Appendices G and H, 10 CFR Part 50,

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Docket Nos. 50-275
and 50-323

Pacific Gas and Electric Company
ATTN: Mr. John C. Morrissey
Vice President & General Counsel
77 Beale Street
San Francisco, California 94106

Gentlemen:

SUBJECT: DIESEL GENERATOR OPERATING STATUS INDICATION
DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2

In our review of reports submitted by licensees concerning malfunctions of diesel generators, we find that in some cases the information available to the control room operator to indicate the operational status of the diesel generator may be imprecise and could lead to misinterpretation. This can be caused by the sharing of a single annunciator station by alarms that indicate conditions that render a diesel generator unable to respond to an automatic emergency start signal and alarms that only indicate a warning of abnormal, but not disabling, conditions. Another cause can be the use of wording on an annunciator window that does not specifically say that a diesel generator is inoperable (i.e., unable at the time to respond to an automatic emergency start signal) when in fact it is inoperable for that purpose.

We, therefore, request that you review the alarm circuitry and diesel generator control circuitry for the diesel generators at your facility to determine how each condition that renders a diesel generator unable to respond to an automatic emergency start signal is alarmed in the control room. These conditions include not only the trips that lock out the diesel generator start and require manual reset, but also control switch or mode switch positions that block automatic start, loss of control voltage, insufficient starting air pressure of battery voltage, etc. This review should consider all aspects of possible diesel generator operational conditions, for example test conditions and operation from local control stations. One area of particular concern is the unreset condition following a manual stop at the local station which terminates a diesel generator test prior to resetting the diesel generator controls for enabling subsequent automatic operation.

Please respond within 45 days of your receipt of this letter by providing the following information:

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(a) all conditions that render the diesel generator incapable of responding to an automatic emergency start signal as discussed above;

(b) the wording on the annunicator window in the control room that is alarmed for each of the conditions identified in (a);

(c) any other alarm signals that also cause the same annunicator to alarm;

(d) any condition that renders the diesel generator incapable of responding to an automatic emergency start signal which is not alarmed in the control room; and

(e) any proposed modifications resulting from this evaluation.

Sincerely,

Original Signed by
John F. Stolz

John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management

cc: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

November 8, 1977

Docket No. 50-275 & 50-323

Pacific Gas and Electric Company
ATTN: Mr. John C. Morrissey
Vice President and Gneeral
Counsel
77 Beale Street
San Francisco, California 94106

Subject:
TRANSMITTAL OF MEETING NOTICE FOR MEETING ON NOVEMBER 15, 1977

The following documents concerning our review of the subject facility are transmitted for your information:

- [ ] Notice of Receipt of Application.
- [ ] Draft/Final Environmental Statement, dated ______.
- [ ] Safety Evaluation, or Supplement No. ____, dated ______.
- [ ] Notice of Hearing on Application for Construction Permit.
- [ ] Notice of Consideration of Issuance of Facility Operating License.
- [ ] Application and Safety Analysis Report, Vol. ______.
- [ ] Amendment No. ______ to Application/SAR, dated ______.
- [ ] Construction Permit No. CPPR- ______, dated ______.
- [ ] Facility Operating License No. DPR-______, NPF-______, dated ______.
- [ ] Amendment No. ____ to CPPR-____ or DRR-____, dated ______.
- [x] Other: ~~Transmittal of meeting notice for meeting on November~~ 15, 1977 with California State Historic Preservation Officer, State of
- [ ] ~~Californaa Planning and Research, and the Ghumash Indi~~an People.

Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: Service List

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