JUL 2 2 1977

DOCKET NOS. 50-361
AND 50-362

APPLICANTS: SOUTHERN CALIFORNIA EDISON COMPANY
SAN DIEGO GAS AND ELECTRIC COMPANY

FACILITY: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

SUBJECT: SUMMARY OF RECENT TRIP TO SAN ONOFRE SITE

Enclosed is a memorandum summarizing a recent trip made by members of the NRC staff to inspect apparent faulting in the vicinity of the San Onofre site.

Original Signed by

Harry Rood
Light Water Reactors
Branch No. 2
Division of Project Management

Enclosure:
Memorandum Summarizing Trip
Dated June 14, 1977

ccs w/enclosure:
See page 2

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GO

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SURNAME ➤	HRood:mt	KKniel				
DATE ➤	7/ /77	7/ /77				

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

JUN 1 4 1977

MEMORANDUM FOR: W. P. Gammill, Assistant Director for Site Technology, DSE

FROM: A. T. Cardone, Geosciences Branch, DSE

THRU: J. C. Stepp, Chief, Geosciences Branch, DSE
R. B. Hofmann, Leader, GSS, GB, DSE

SUBJECT: MAY 25 AND 26, 1977 SITE VISIT TO THE SAN ONOFRE NUCLEAR STATION - DOCKET NO. 50-361/362

On May 25 and 26, 1977 R. Jackson and I visited the area a few miles south of the San Onofre Nuclear Station, where apparent faulting was reported on May 20, 1977 by the staff of the California Energy Resources Conservation and Development Commission (CERCDC). We were accompanied by representatives of the Southern California Edison Company (SCEC) and their FUGRO consultants, CERCDC, ACRS staff and consultants, and the U.S. Corps of Engineers. An attendance list is enclosed.

For approximately two hours in the late afternoon of May 25, the staff, accompanied by SCEC representatives and consultants, made a preliminary inspection of the area of reported apparent faulting. On the morning of May 26 the entire group of representatives from the above agencies met at the San Onofre Nuclear Station visitor's center. After a few introductory comments by SCEC, the group visited four locations where terrace deposits have been offset and where faulting in San Onofre breccia had been reported.

The first stop was made approximately 3 miles southeast of the San Onofre plant along the beach, where we were shown photographs with superimposed geologic mapping of the shears offsetting the basal marine terrace deposits, and of the landslide which appears to encompass the terrace deposits. These features were physically inspected by the group.

The second stop was at Horno Canyon where we inspected offsets at the terrace deposit-bedrock contact. They appeared to be associated with a graben-like slumped block behind a massive landslide.

The third stop was at Las Pulgas Canyon where we observed an apparent normal fault dipping approximately 50° East and offsetting the base of the terrace deposits. There was no landsliding associated with this feature.

The fourth and last stop was on Camp Pendleton property at an outcrop of faulted San Onofre breccia of Miocene age. The fault had not been mapped nor dated but it appeared to trend in the direction of the sea cliff shears to the southeast.

After the field inspections a meeting was held in which SCE and their consultants described the following investigatory program being performed:

1. Review of published and unpublished geologic reports.

2. Analysis of specially flown vertical, oblique, and stereo photographs.

3. Study of stratigraphic correlations in the terrace deposits and a geometric reconstruction of the pre-slide configuration of the offset beds.

4. Detailed geologic mapping of the shears and landslide area.

5. Additional mapping and dating of the Camp Pendleton fault.

6. More detailed mapping and photo geology work will be done at Horno and Las Pulgas Canyons.

SCE provided NRC staff single copies of two reports, "Landsliding in Marine Terrace Terrain, California" by George B. Cleveland 1975, CDMG Special Report 119, and "Preliminary Geologic Report of the Coastal Central Portion of Camp Pendleton marine base, San Diego County, California, June 4, 1971, by Converse-Davis and Associates, conducted for the City of Los Angeles Department of Water and Power. These reports address the geology of the field locations inspected by us. In addition, two sets of photos with overlays containing some geologic interpretations were provided to us.

After a caucus between the ACRS consultants, CDMG, and us, SCE was informed that we and the ACRS concurred in the investigatory program that they had undertaken, however, they should contact CDMG for their recommendations. We agreed to keep in close contact with SCE during the program, and to provide timely comments and recommendations. SCE stated that we should receive a final report of investigations in approximately one month.

A. T. Cardone
A. T. Cardone
Geosciences Branch
Division of Site Safety and
Environmental Analysis

Enclosure:
As stated

cc: see attached list

JUN 1 / 1977

ATTENDANCE LIST

Name	Organization
H. Stoehr	SDG&E
G. Thompson	ACRS Consultant
J. Maxwell	ACRS Consultant
G. Quittschreiber	ACRS
L. Knuppel	Corps of Engineers
A. T. Cardone	USNRC
R. Jackson	USNRC
G. K. Lee	CERCDC
R. L. Strand	CERCDC (Energy Commission)
P. Y. Amimoto	CDMG
G. Hunt	SCE
G. Hawkins	SCE
P. West	SCE
D. Nunn	SCE
J. Rainsberry	SCE
J. Haynes	SCE
P. Davis	FUGRO
J. Scott	FUGRO
D. Miller	FUGRO

JUL 22 1977

DOCKET NOS. 50-361
AND 50-362

APPLICANTS: SOUTHERN CALIFORNIA EDISON COMPANY
SAN DIEGO GAS AND ELECTRIC COMPANY

FACILITY: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

SUBJECT: SUMMARY OF MEETING TO DISCUSS METEOROLOGICAL TRACER TESTS

Members of the NRC staff and its consultants met with representatives of the applicants and their consultants in Bethesda, Maryland on May 18, 1977 to discuss the interim results from the meteorological tracer tests being conducted at San Onofre. Attendees at the meeting are given in Enclosure 1. The material presented by the applicants is given in Enclosure 2. The applicants indicated that the final report on the tracer tests would be submitted in the near future, and the FSAR would be modified to reflect the tracer test results in the amendment that provides responses to first round questions. The staff stated that when the second round questions are issued, the staff would take positions on whether or not the bluff tower meteorological data is acceptable and whether or not a site-specific dispersion model based on the tracer tests is acceptable.

Original Signed by

Harry Rood
Light Water Reactors
Branch No. 2
Division of Project Management

Enclosures:
1. Attendance list
2. Material Presented by
the Applicants

ccs w/enclosures:
See page 2

M 4
GP

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SURNAME➤	HRood:mt	KKniel				
DATE➤	7/ /77	7/ /77				

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JUL 2 2 1977

ENCLOSURE 1

ATTENDANCE LIST
MEETING WITH SOUTHERN CALIFORNIA EDISON COMPANY AND
SAN DIEGO GAS AND ELECTRIC COMPANY
SAN ONOFRE
MAY 18, 1977

NRC - STAFF

H. Rood
E. Markee
M. Rubin
Millard Wohl
A. Burger
John Goll

SCECo.

W. C. Moody
L. J. Brunton

SAI

L. H. Teuscher

NUS

J. H. Taylor
E. Mitchell
M. Septoff

NOAA

I. VanderHoven

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SURNAME➤						
DATE➤						

ENCLOSURE 2

SAN ONOFRE METEOROLOGICAL TRACER TESTS
MEETING AGENDA
May 18, 1977

I. Objectives of Program

- A. Characterize Dispersion at SONGS for Accident Conditions
- B. Characterize Dispersion at SONGS for Routine Releases
- C. Evaluate Appropriateness of Using Bluff Tower Meteorology for Estimating SONGS Dispersion

II. Program Description

- A. Tracer and Meteorological Monitoring
- B. Meteorological Conditions

III. Interim Results

- A. Test Condition Meteorology
- B. $\chi\overline{u}/Q$
- C. Comparison of Observed vs. Calculated χ/Q

IV. Interim Conclusions

- A. Little Difference Between Inland and Bluff Tower to Characterize Dispersion
- B. Observed Peak χ/Q's Lower for Neutral and Slightly Stable Conditions

V. Implications of Tracer Tests on Design Basis Accident Meteorology for Units 2 and 3

VI. Future Submittals

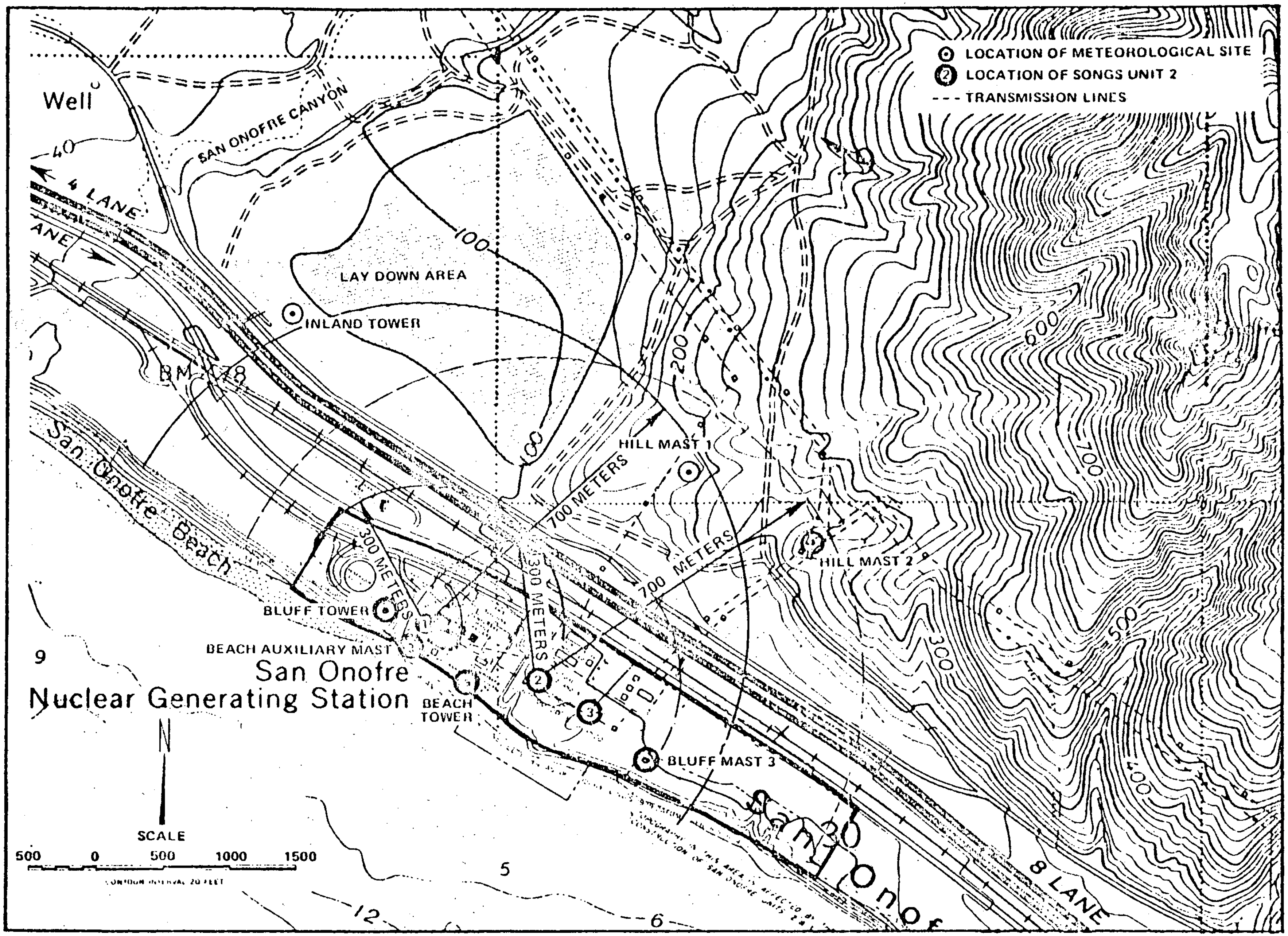
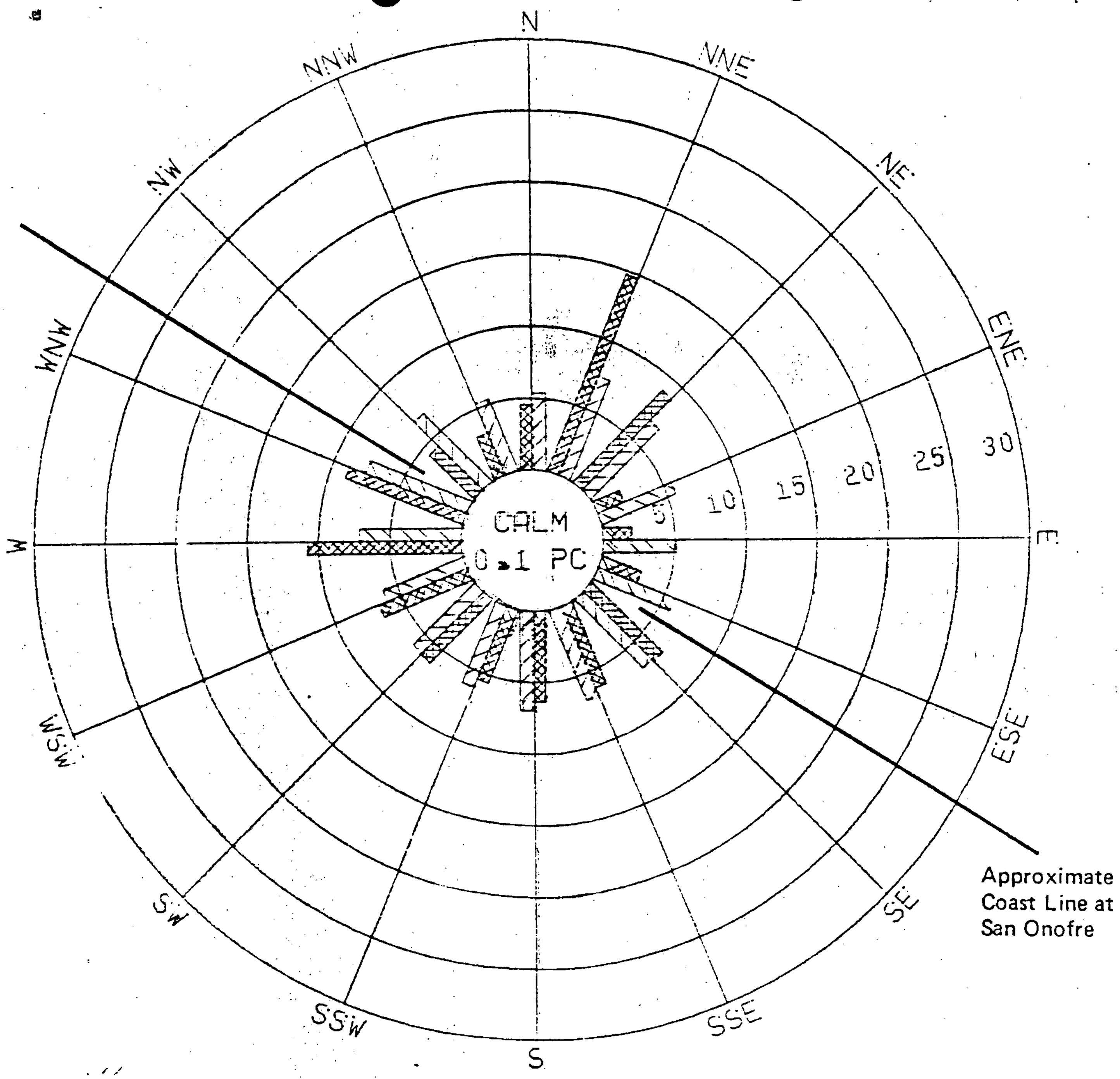


FIGURE 2.1-2 SONGS TERRAIN AND APPROXIMATE LOCATIONS OF METEROLOGICAL SITES AND SAMPLER ARCS

4/77



WIND DIRECTION FREQUENCY (PERCENT)

MEAN WIND SPEED (MPH)

FIGURE 2.2-1

ANNUAL WIND ROSE FOR SONGS BASED ON
THE 10 METER LEVEL BLUFF TOWER (1/25/75 – 1/24/76)

TABLE 2.2-2

FREQUENCY (%) OF OCCURRENCE OF DILUTION CLASSES AT SONGS FOR ONSHORE FLOW

The values (from SCE, 1976) are based on the main bluff tower data, 10 m winds and ΔT 120ft-20ft, for the period of February 1, 1975 to January 28, 1976. Winter includes the months of December, January, and February. The frequency, rounded to the nearest whole percent, is based on the total number of observations for all directions during the time period. Values may not necessarily total exactly due to rounding. Onshore flow includes the directions of southeast clockwise through west-northwest.

Dilution Class	Time Period: Annual	Time Period: Winter
Most restrictive	4%	8%
Moderately restrictive	10%	10%
Least restrictive	44%	26%
All classes	58%	45%

TABLE 4.1-1

WINTER PERCENT FREQUENCIES BASED ON TOTAL OBSERVATIONS OF PASQUILL STABILITY CLASS DURING ONSHORE FLOW AT SONGS

The values for the period December 1, 1976, to February 28, 1977, are based on 10-m winds and ΔT(40m-10m) from the main Bluff Tower. The values for the period January 25, 1973, to January 24, 1976, are based on 10-m winds and ΔT(36.5m-10m) from the main Bluff Tower. Winter consists of the months December, January, and February. The frequency is based on the total number of observations for all directions during the time period and is rounded to the nearest whole percent. Onshore flow includes the wind direction of southeast clockwise through west-northwest Values may not necessarily total exactly due to rounding.

Winter Period	Pasquill Stability Class A	B	C	D	E	F	G	Total*
Onshore Program (12/1/76-2/28/77)	16	2	1	7	2	1	1	30
Prior Three Years (1/25/73-1/24/76)	12	1	1	11	9	4	4	42

*Total represents the percent frequency of all onshore flow for the period.

TABLE 4.1-2

PERCENT FREQUENCY OF OCCURRENCE BASED ON TOTAL OBSERVATIONS OF WINTER DILUTION/DIRECTION CLASS AT SONGS

The values for the winter period 12/76-2/77 are based on 10-m winds and ΔT(40m-10m) from the main bluff tower. The values for the winter period 1/73-1/76 are based on 10-m winds and ΔT(120ft-20ft) from the main Bluff Tower. Winter consists of the months of December, January, and February. Values may not total exactly due to rounding. The two-digit number in the upper-left corner of each blank is the dilution/direction class.

Dilution Potential Class	Wind Direction Class			Sub-Total
	(1) Alongshore Flow-Southerly (SE-SSE) ($130^{\circ}-169^{\circ}$)	(2) Direct Onshore Flow (S,SSW,SW,WSW) ($170^{\circ}-259^{\circ}$)	(3) Alongshore Flow-Northerly (W-WNW) ($260^{\circ}-300^{\circ}$)	
(1) Least Restrictive	11 onshore program 3 three-year 7	12 onshore program 8 three-year 10	13 onshore program 13 three-year 9	24 25
(2) Moderately Restrictive	21 onshore program 1 three-year 4	22 onshore program 2 three-year 3	23 onshore program 2 three-year 3	5 10
(3) Most Restrictive	31 onshore program 0 three-year 3	32 onshore program 0 three-year 2	33 onshore program 0 three-year 1	1 6

TABLE 4.1-4

NUMBER OF ONSHORE TRACER TESTS BY DILUTION/DIRECTION CLASS

The following are the number of onshore tests in the interim report for SONGS for the period of December 1976 through February 1977. The two digit number in the upper-left corner of each block is the dilution/direction class.

Dilution Potential Class	Wind Direction Class			Sub-Total
	(1) Alongshore Flow-Southerly (SE-SSE) (130°-169°)	(2) Direct Onshore Flow (S, SSW, SW, WSW) (170°-259°)	(3) Alongshore Flow-Northerly (W-WNW) (260°-300°)	
(1) Least Restrictive	11 1	12 11	13 14	26
(2) Moderately Restrictive	21 1	22 5	23 2	8
(3) Most Restrictive	31	32	33	0
Subtotal	2	16	16	34

TOTAL TESTS: 34

TABLE 4.1-6

CONTINGENCY TABLE OF ΔT STABILITY CLASS - BLUFF TOWER VERSUS INLAND TOWER

The Table shows the number of occurrences of stability classes based on 1-h average ΔT measurements during each test. Presented are the tests included in the interim report of the Onshore Tracer Program at SONGS.

Bluff Tower $\Delta T_{40m-10m}$ Stability Class	Inland Tower $\Delta T_{40m-10m}$ Stability Class A	B	C	D	E	F	G	TOTAL
A	9	4		6				19
B								
C	1	1		1				3
D	2	2	3	5				12
E								
F								
G								
TOTAL	12	7	3	12				34

TABLE 4.1-7

CONTINGENCY TABLE OF σ_θ STABILITY CLASS - BLUFF TOWER VERSUS INLAND TOWER

Table shows the number of occurrences of stability classes based on the indicated 1-h average σ_θ calculations for each test. Presented are the tests included in the interim report of the Onshore Tracer Program at SONGS.

Inland Tower 10 m σ_θ Stability Class (columns); Bluff Tower 10 m σ_θ Stability Class (rows)

	A	B	C	D	E	F	G	TOTAL
A	2	2						4
B	1	3	2					6
C		4		1				5
D	1	1	6	2	2			12
E			1	5	1			7
F								
G								
TOTAL	4	10	9	8	3			34

TABLE 4.1-5

NUMBER OF OCCURRENCES OF PASQUILL STABILITY CLASS DURING THE ONSHORE PROGRAM TESTS AT SONGS

Stability classes were determined by the Bluff Tower. All test conditions were for onshore flow.

Stability Classifier	Pasquill Stability Class A	B	C	D	E	F	G	Total
ΔT(40m-10m)	19	0	3	12	0	0	0	34
10-m σ_θ	4	6	5	12	7	0	0	34

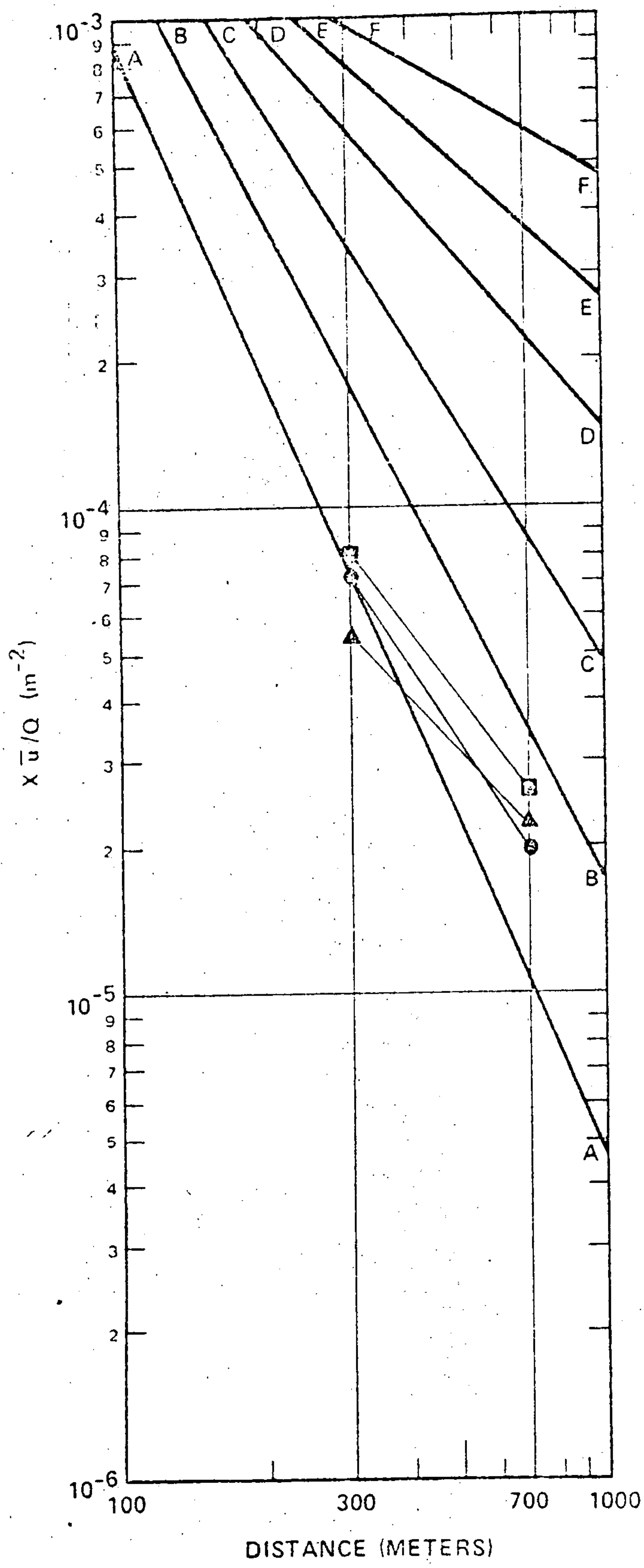


FIGURE 4.2-1

MEAN PEAK $\chi \bar{u}/Q$ VERSUS DOWNWIND DISTANCE BY STABILITY FOR BLUFF TOWER ΔT

The lines labeled with symbols (see key below for stability class represented) represent the mean for the test data. The mean was determined for each stability class by using the individual values and assuming a nominal downwind distance of 300 or 700 m as appropriate. The individual values were based on observed tracer materials, the 10 m wind speed, and ΔT(40-10m) stability class. The lines labeled with letters are for orientation and they include the mean effect of the building wake factors of Units 1 and 2.

KEY

A — ⊡
B — □
C — ▲
D — ⊙
E — ○

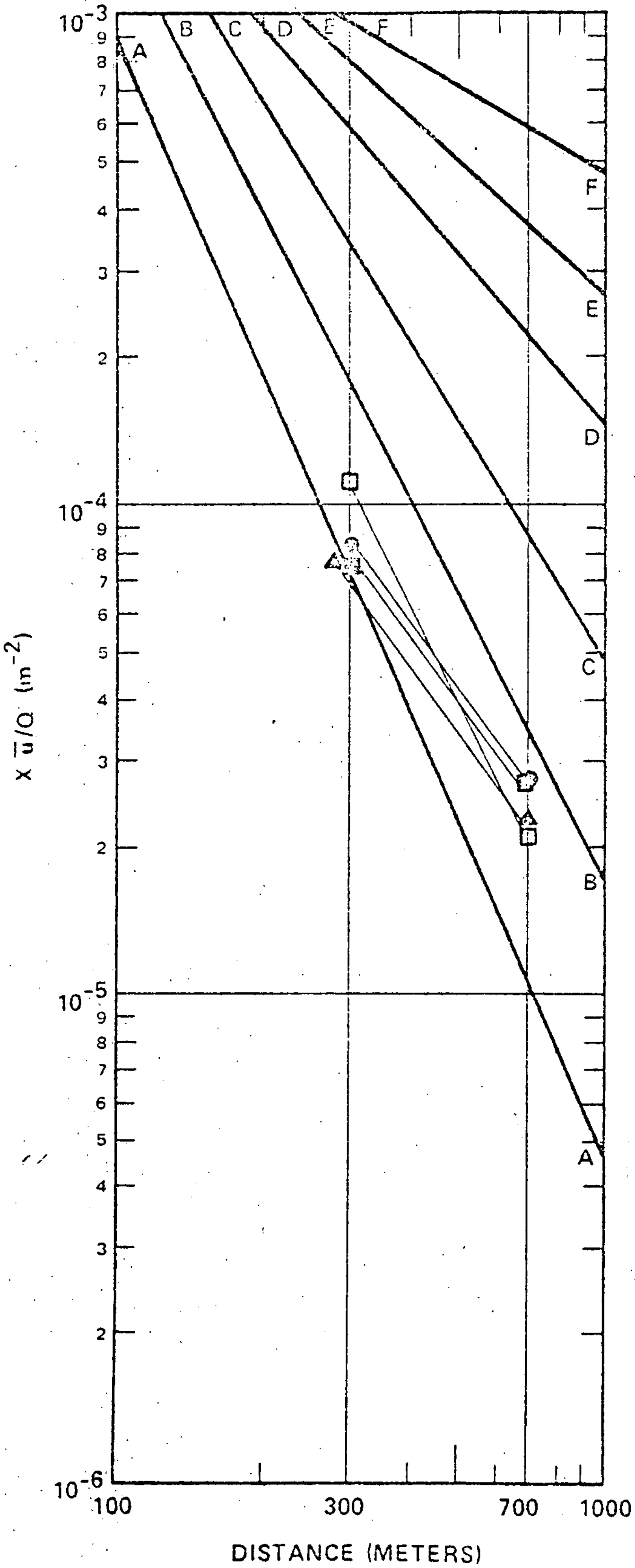


FIGURE 4.2-3

MEAN PEAK $\chi\ \bar{u}/Q$ VERSUS DOWNWIND DISTANCE BY STABILITY FOR INLAND TOWER ΔT

The lines labeled with symbols (see key below for stability class represented) represent the mean for the test data. The mean was determined for each stability class by using the individual values and assuming a nominal downwind distance of 300 or 700 m as appropriate. The individual values were based on observed tracer materials, the 10 m wind speed, and ΔT(40-10m) stability class. The lines labeled with letters are for orientation and they include the mean effect of the building wake factors of Units 1 and 2.

KEY

A — ◘

B — □

C — ▲

D — ●

E — ○

FIGURE 4.3-1

OBSERVED VERSUS CALCULATED χ/Q FOR STABILITY CLASSES A, B, C BASED ON BLUFF TOWER ΔT(40-10m)

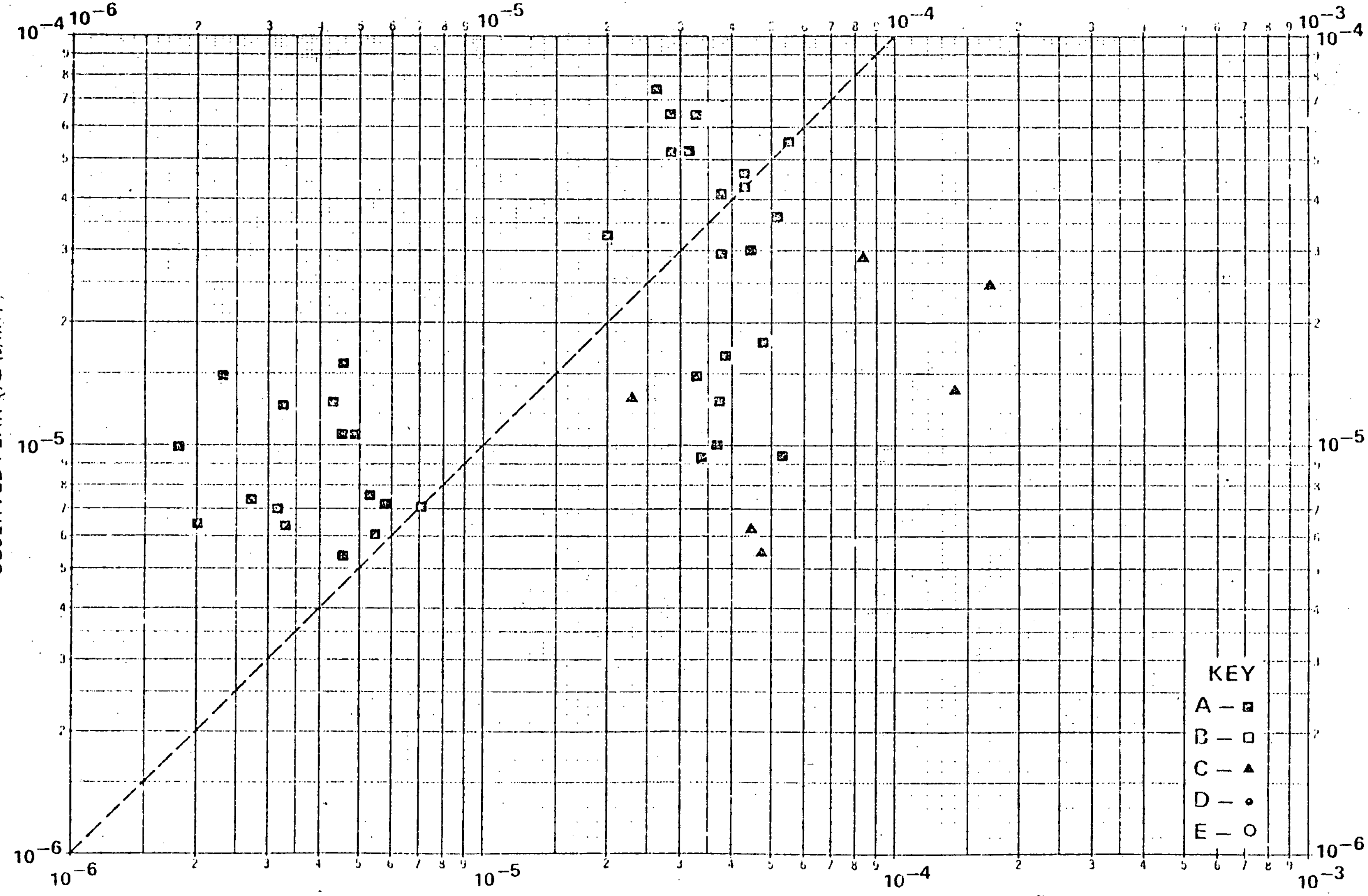


FIGURE 4.3–2

OBSERVED VERSUS CALCULATED χ/Q FOR STABILITY CLASS D BASED ON BLUFF TOWER ΔT(40-10m)

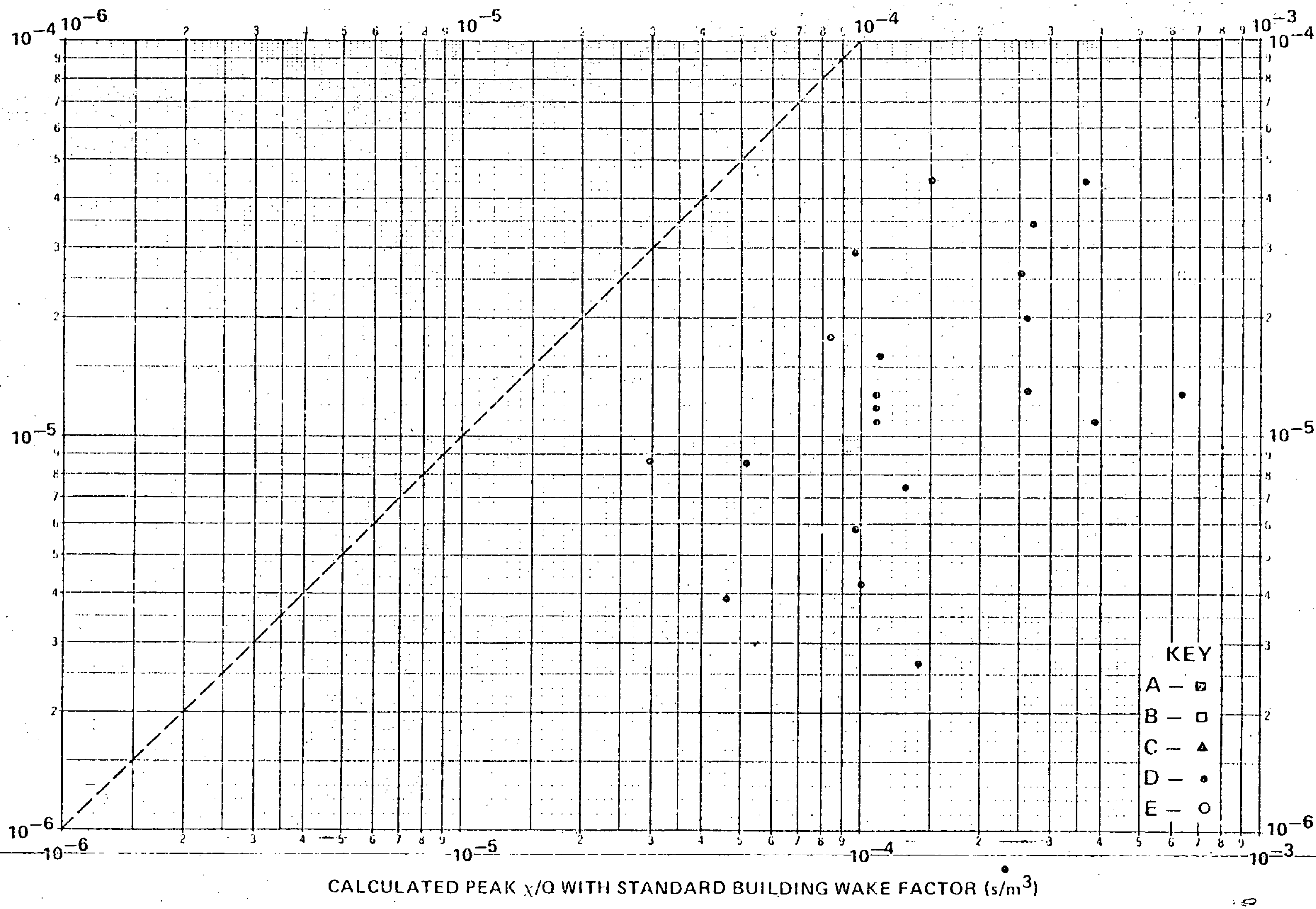


FIGURE 4.3–3

OBSERVED VERSUS CALCULATED χ/Q FOR STABILITY CLASSES A, B, C BASED ON INLAND TOWER ΔT(40-10m)

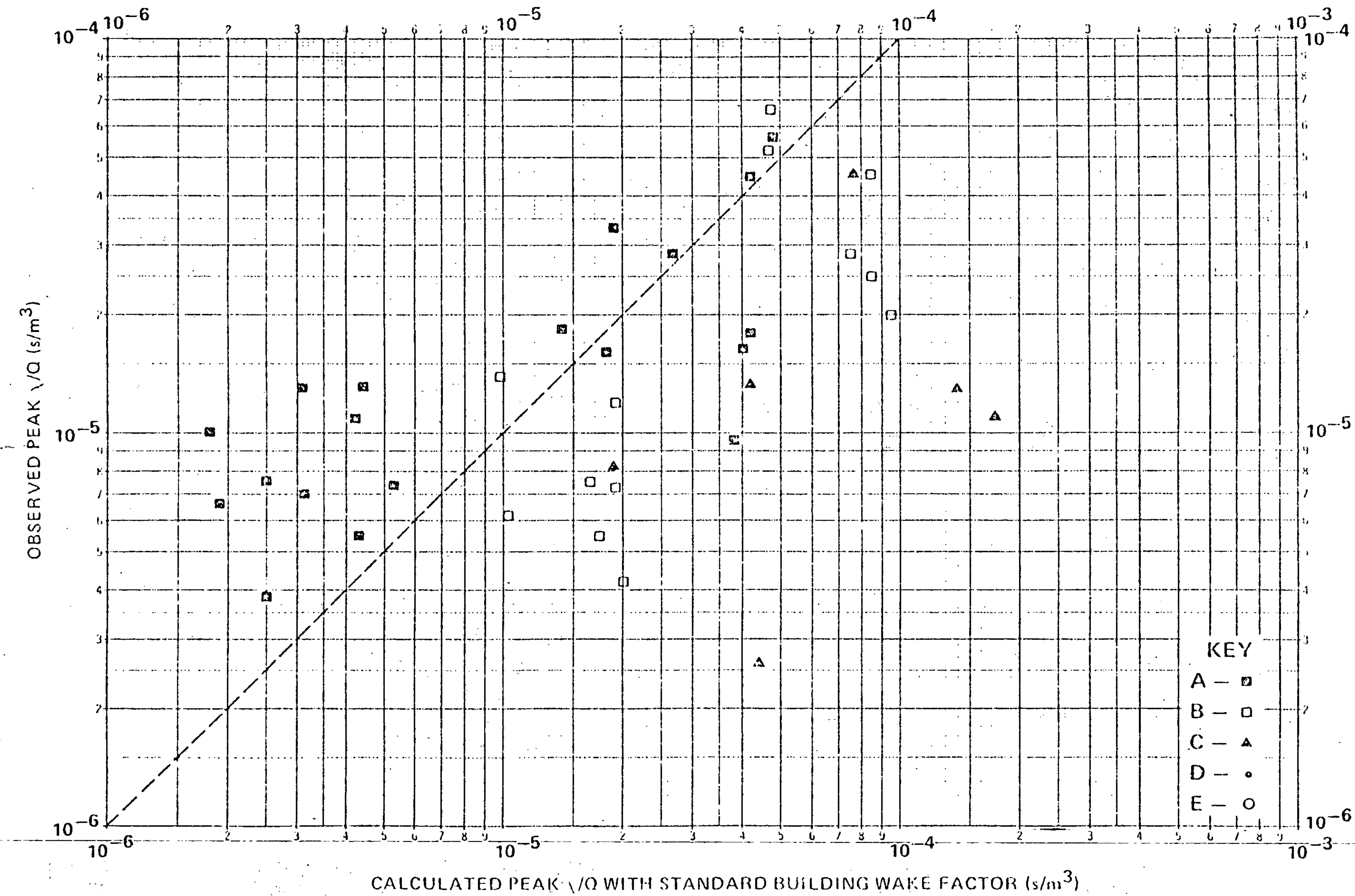


FIGURE 4.3–4

OBSERVED VERSUS CALCULATED χ/Q FOR STABILITY CLASSES D, E BASED ON INLAND TOWER ΔT(40-10m)

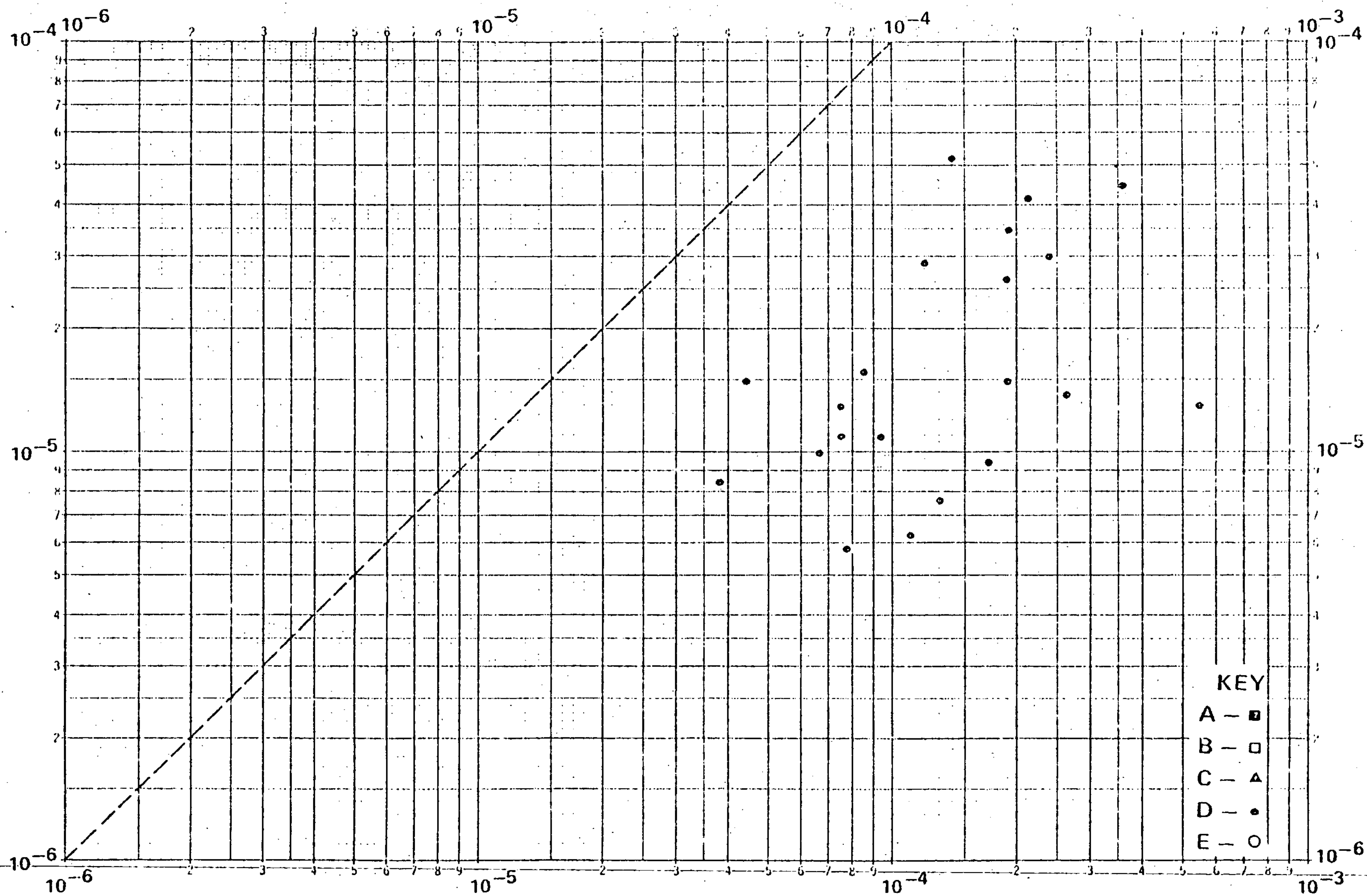


TABLE 4.3-9

MEDIAN RATIO BY STABILITY CLASS OF OBSERVED VERSUS CALCULATED PEAK χ/Q BASED ON ΔT(40m-10m)

The stability classifications were based on 1-h averages during the test. The calculated values included the standard building wake factor. N/A means "none available" observations for that stability class.

a) Bluff Tower Meteorology

Stability Class	Median Ratio of Calculated/Observed
A	0.7
B	N/A
C	7.0
D	11.1

b) Inland Tower Meteorology

Stability Class	Median Ratio of Calculated/Observed
A	0.7
B	2.3
C	7.1
D	7.5

TABLE 4.3-20

MEDIAN RATIO BY STABILITY CLASS OF OBSERVED VERSUS CALCULATED PEAK χ/Q BASED ON 10-m σ_θ MEASUREMENTS

The stability classifications were based on 1-h averages during the test. The calculated values included the standard building wake factor. N/A means "none available" observations for that stability class.

a) Bluff Tower Meteorology

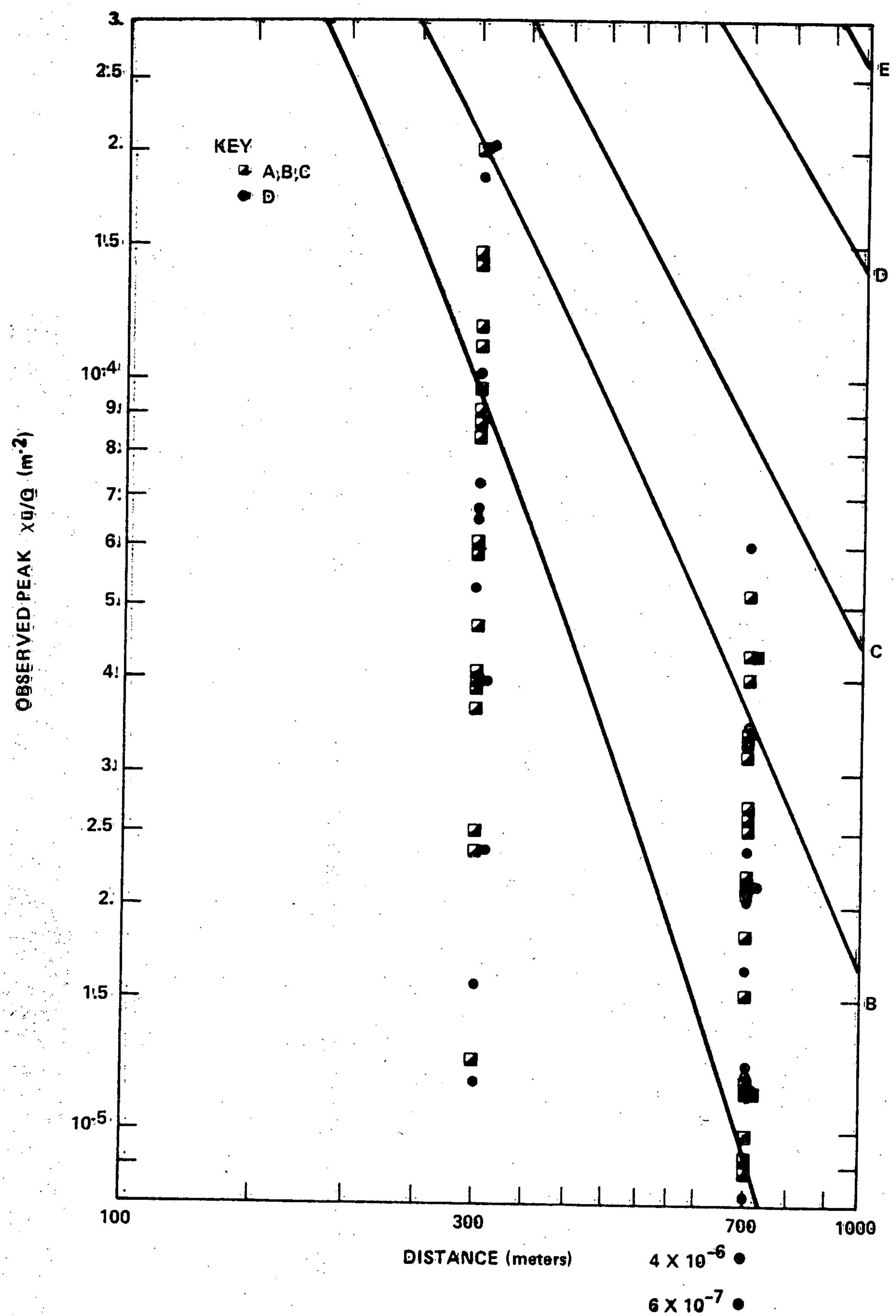
Stability Class	Median Ratio of Calculated/Observed
A	1.2
B	4.3
C	4.7
D	6.9
E	8.3

b) Inland Tower Meteorology

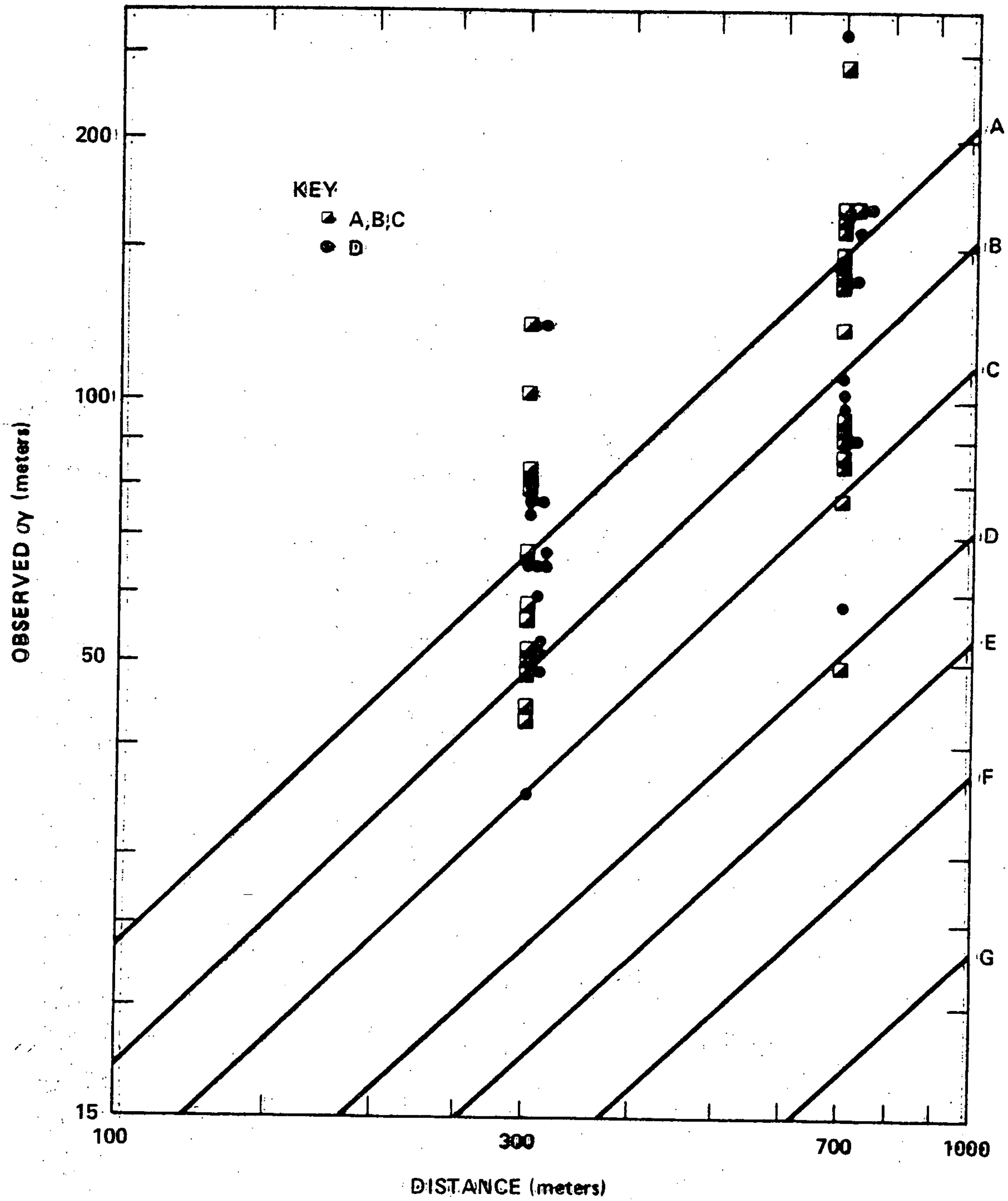
Stability Class	Median Ratio of Calculated/Observed
A	0.9
B	2.7
C	2.7
D	7.4
E	5.0

INTERIM CONCLUSIONS FOR SONGS ONSHORE TRACER PROGRAM

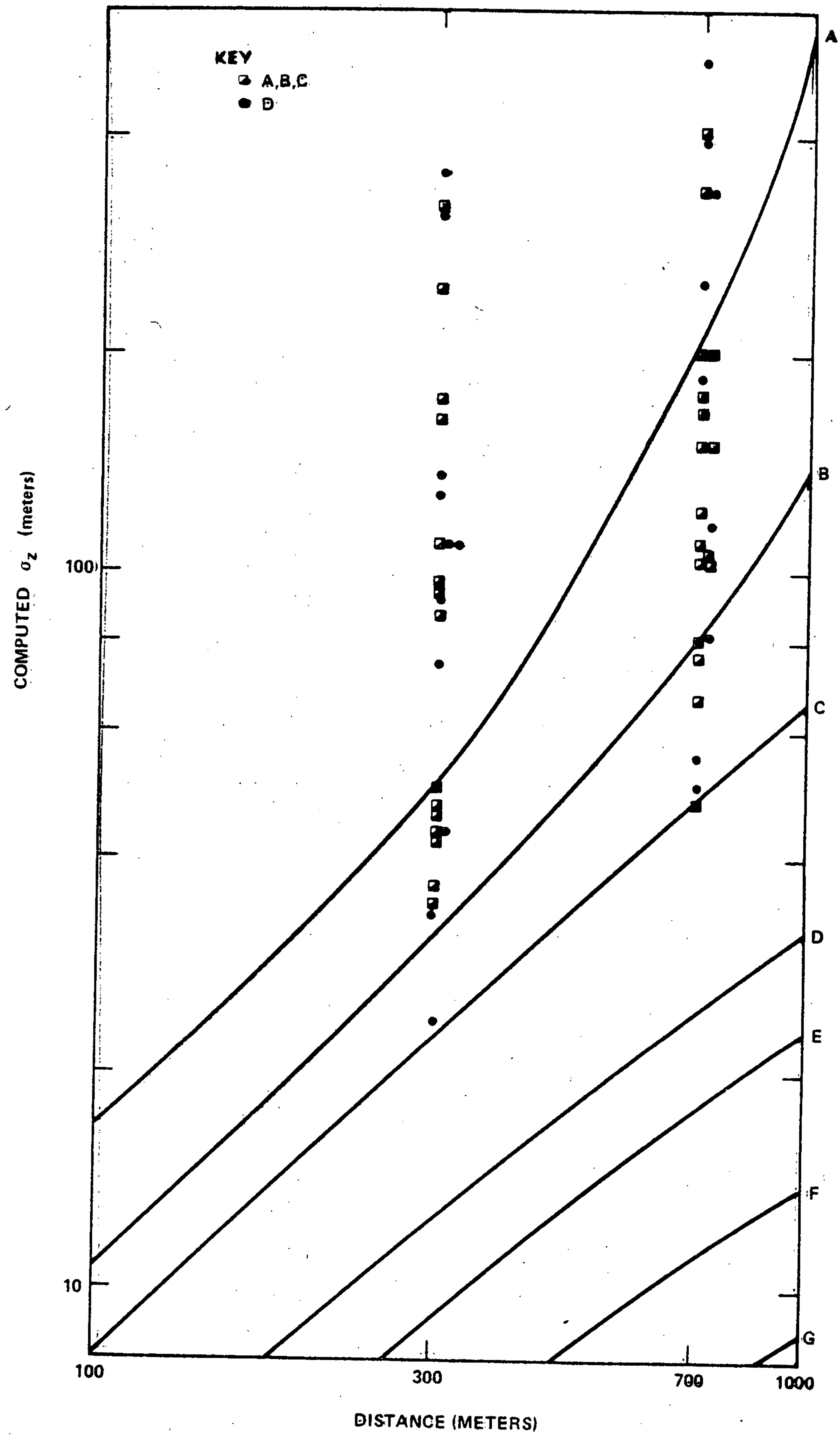
- LITTLE DIFFERENCE BETWEEN BLUFF AND INLAND TOWERS TO CHARACTERIZE DISPERSION
- OBSERVED PEAK χ/Q'S LOWER FOR NEUTRAL AND SLIGHTLY STABLE CONDITIONS



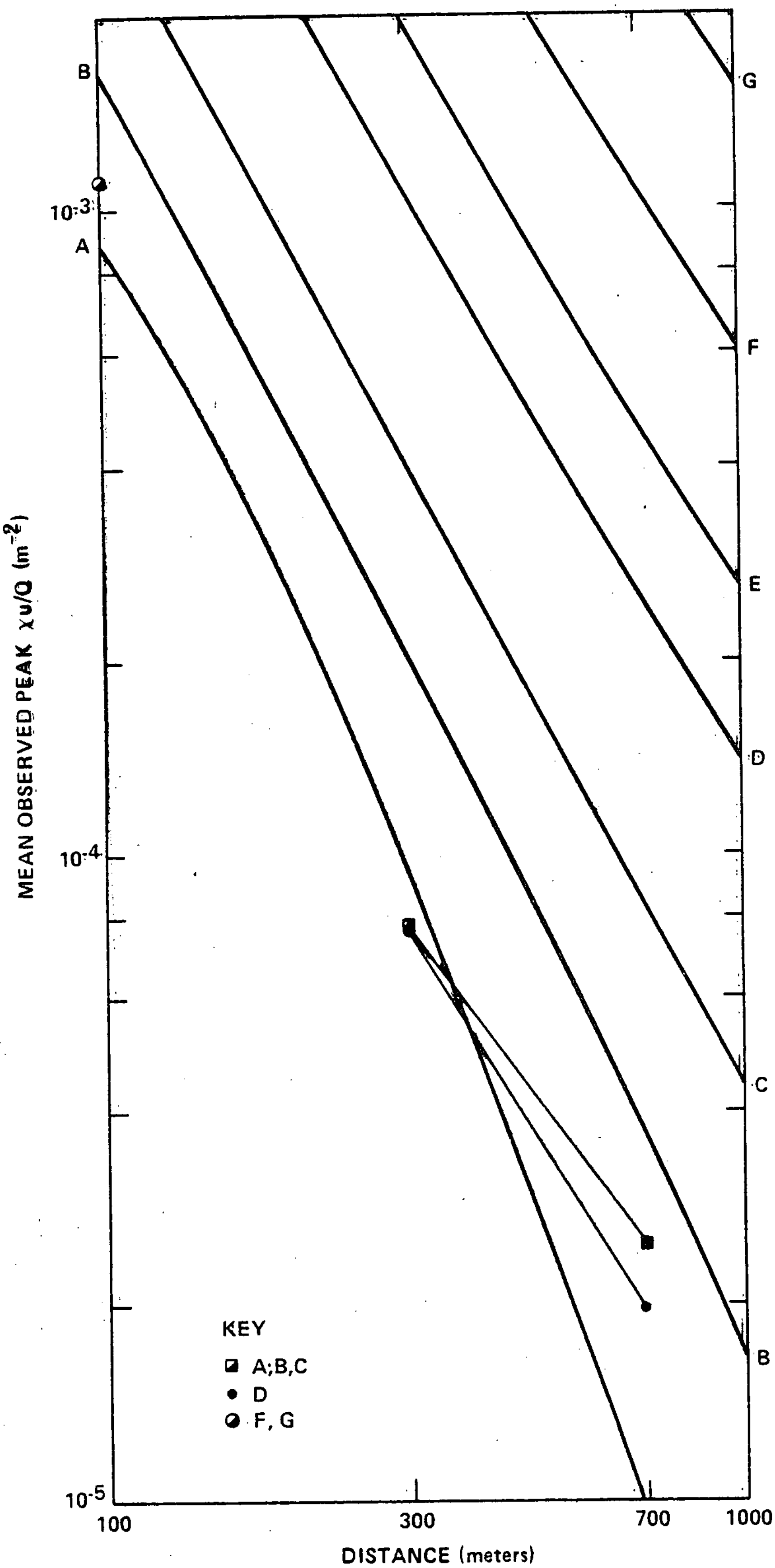
OBSERVED PEAK $\chi\bar{u}/Q$ vs DISTANCE



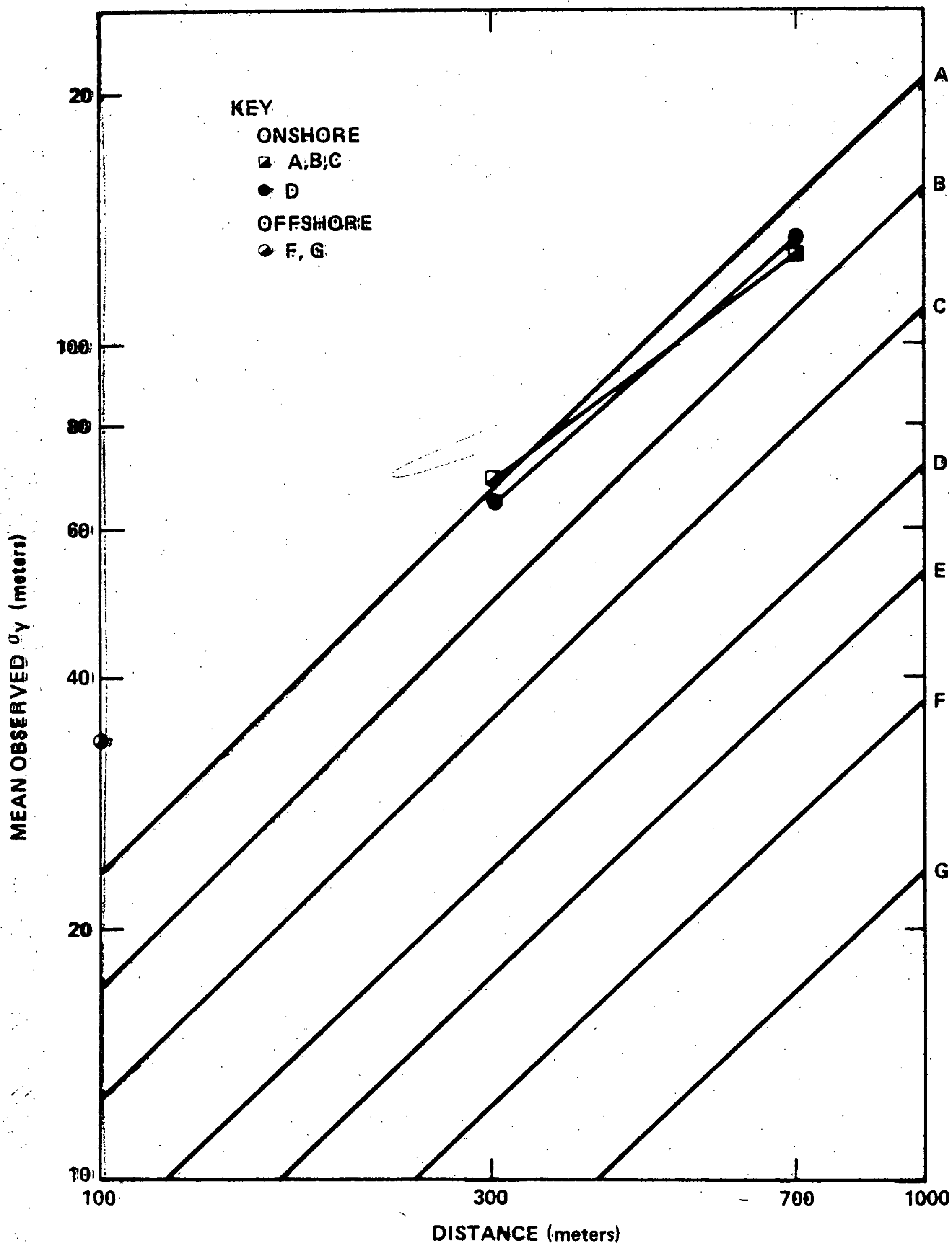
OBSERVED σ_y vs DISTANCE



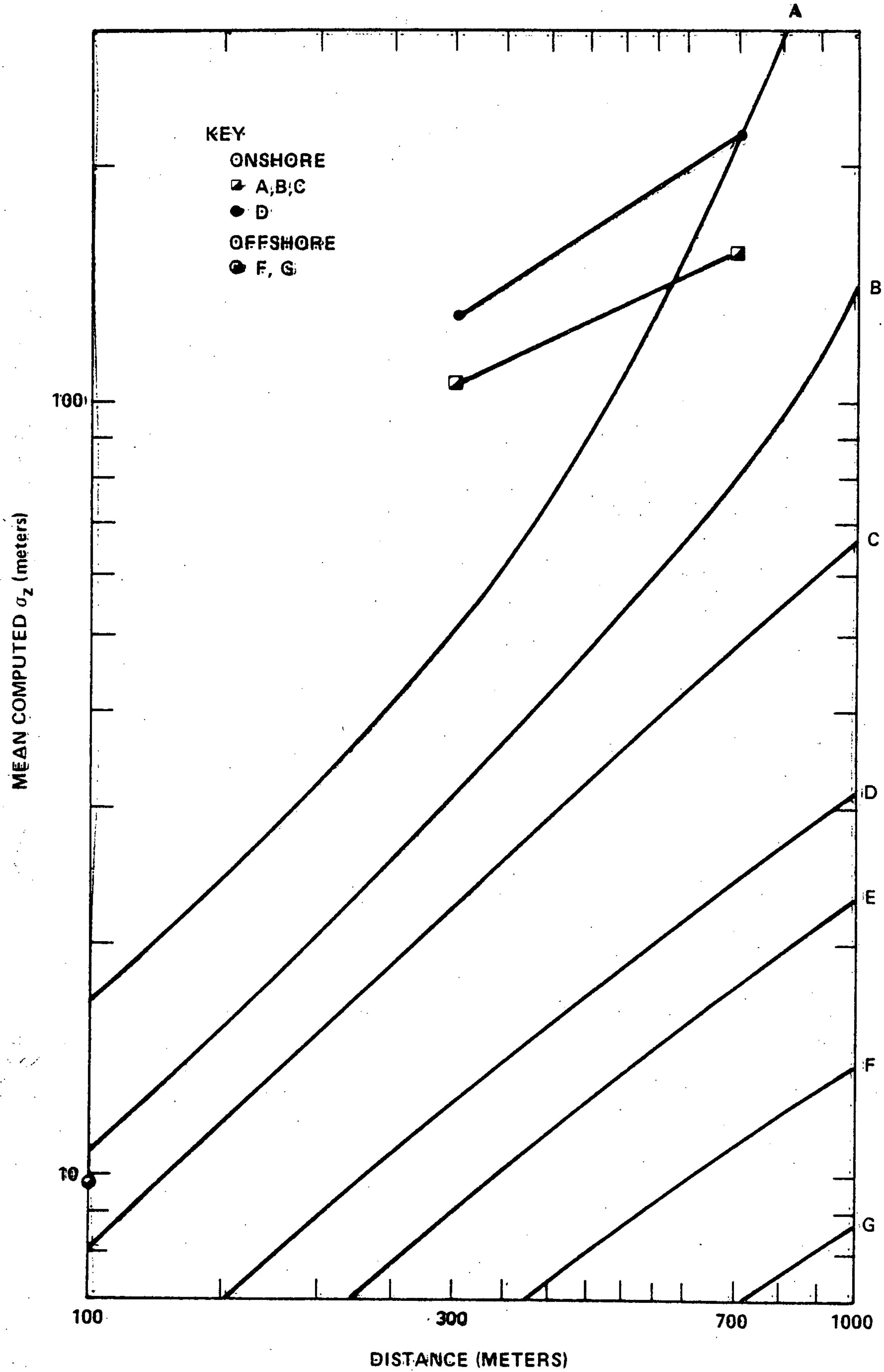
COMPUTED σ_z vs DISTANCE



MEAN OBSERVED PEAK $\chi u/Q$ vs DISTANCE



MEAN OBSERVED σ_y vs DISTANCE



MEAN COMPUTED σ_z vs DISTANCE

SONGS

DISPERSION PARAMETERS AT 576 METERS

Stability Class	Pasquill-Gifford (m) σ_y	Pasquill-Gifford (m) σ_z	Adjustment Factor σ_y	Adjustment Factor σ_z	Adjusted Parameter (m) Σ_y	Adjusted Parameter (m) Σ_z	R.G.(1) $\chi\bar{u}/Q$ ($\times 10^{-4}$) (m^{-2})	SONGS(2) $\chi\bar{u}/Q$ ($\times 10^{-4}$) (m^{-2})	R(3)
A	124	151	0.8	0.6	99	91	.167	.353	.5
B	91	63	1.0	1.2	91	75	.521	.467	1.1
C	66	40	1.3	1.5	86	60	1.07	.618	1.7
D	43	21	2.0	2.6	83	55	2.56	.697	3.7
E	32	15	2.5	3.5	80	52	3.81	.765	5.0
F	23	10	3.0	5.0	69	50	5.75	.926	6.2
G	14	6	4.0	8.0	56	48	12.6	1.18	10.7

(1) $$\text{R. G. } \chi\bar{u}/Q = \left[\pi\ \sigma_y\ \sigma_z + cA\right]^{-1} \geq \left[3\ \pi\ \sigma_y\ \sigma_z\right]^{-1}$$

(2) $$\text{SONGS } \chi u/Q = \left[\pi\ \Sigma_y\ \Sigma_z\right]^{-1}$$

(3) $$R = \frac{\text{R.G. } \chi\bar{u}/Q}{\text{SONGS } \chi\bar{u}/Q}$$

JUN 2 4 1977

Docket Nos. 50-~~300~~ 361
50-~~361~~ 362

MEMORANDUM FOR: G. S. Spencer, Chief, Reactor Construction and Engineering Support Branch, RV

FROM: G. W. Reinmuth, Assistant Director, Division of Reactor Construction Inspection, IE

SUBJECT: SAN ONOFRE UNITS NOS. 2 AND 3
10 CFR 50.55(e) LETTER DATED MAY 27, 1977

We have reviewed the report transmitted as an enclosure to the subject letter, and have the following comments.

The report describes an inspection system designed to verify that all welding on structural steel meets acceptance criteria detailed in Appendix A to the report. These criteria appear to be generally appropriate, with the exception of 3.1.4. In our opinion, "roll over" should not be permitted in Category A joint welds.

The report is silent on the bases for acceptance of the 300 welds identified (page 3) as inaccessible for inspection. We suggest that the licensee be asked to provide a discussion on this subject.

Since the report and its Appendix represent a change from commitments in the PSAR, we are forwarding copies to NRR:DPM and DSS, for their information and review.

G. W. Reinmuth, Assistant Director
Division of Reactor Construction
Inspection, IE

cc w/incoming: (ltr dtd 5/27/77 from Southern California Edison Company to RV, atten. R. H. Engelken, Director).
D. B. Vassallo, DPM
J. P. Knight, DSS

Distribution:
IE Files RCI Reading Central Files (2)
IE Reading RCI Chron JB Henderson

OFFICE➤	RCI:IE	A/D RCI:IE				
SURNAME➤	JB Henderson	GW Reinmuth				
DATE➤	6/ /77 cd	6/ /77				

Southern California Edison Company

P. O. BOX 800

2244 WALNUT GROVE AVENUE

ROSEMEAD, CALIFORNIA 91770

JACK B. MOORE
VICE PRESIDENT

TELEPHONE
213-572-2292

May 27, 1977

Office of Inspection and Enforcement
Region V
U. S. Nuclear Regulatory Commission
Suite 202
1990 North California Boulevard
Walnut Creek, California 94596

Attention: Mr. R. H. Engelken, Director

Gentlemen:

Subject: Docket Nos. 50-361 and 50-362
San Onofre Nuclear Generating Station, Units 2 and 3

This letter forwards our final report pursuant to 10CFR50.55(e) concerning deficiencies identified in structural steel and miscellaneous metal welding at San Onofre Units 2 and 3. An interim report on this subject was provided by my letter dated April 14, 1977.

Enclosed are twenty-five copies of a document entitled, "Final Report on Deficiencies in Welding of Structural Steel and Miscellaneous Metal, San Onofre Nuclear Generating Station, Units 2 and 3." If you have any questions concerning this matter, or if you would like additional information, please let me know.

Very truly yours,

JBMoore

Enclosures
cc: Dr. Ernst Volgenau (NRC, Director I&E)

FINAL REPORT ON DEFICIENCIES IN WELDING OF STRUCTURAL STEEL AND MISCELLANEOUS METAL
SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

INTRODUCTION

This final report is submitted pursuant to 10CFR50.55(e)(3) and it describes the program instituted to verify the adequacy of welding of structural steel and miscellaneous metal performed at the San Onofre Units 2 and 3 construction site.

BACKGROUND

In a letter to the Nuclear Regulatory Commission (NRC) Region V office dated March 15, 1977, Southern California Edison (SCE) confirmed notification of a condition involving welding of structural steel and miscellaneous metal which was thought to be reportable in accordance with 10CFR50.55(e). This was followed by an interim report dated April 14, 1977, which described the status of investigation of the condition and action initiated to resolve deficiencies which were identified.

DISCUSSION

The following discussion is responsive to 10CFR50.55(e)(3).

Description of the Deficiency

Scope —

The design basis criteria stated in the Safety Analysis Report for the subject Units establishes that the design, fabrication and installation of structural steel and miscellaneous metal shall comply with the requirements of the American Institute of Steel Construction (AISC) and the American Welding Society (AWS) structural Welding Code D1.1-1972. Structural steel includes all primary load carrying components of the building framing system while miscellaneous metal includes embedments, grating, deck plate, pipe supports (excluding ASME Boiler and Pressure Vessel Code Section III, Division 1 application), pipe whip restraints (for pipe sizes less than 2-1/2 inches), electrical raceway supports, instrument supports, duct supports and metal work associated with mechanical and electrical equipment supports, excluding sheet metal applications which are not covered by AWS requirements.

Work to these requirements performed in the field by Bechtel, and work to similar requirements imposed via a procurement specification and performed by University Mechanical Engineering Contractors (UMEC), is included in the scope of this report.

Deficient Conditions —

Review of Bechtel and UMEC work by Bechtel identified a failure to consistently comply with all of the visual inspection criteria specified. The deficiencies included both insufficient and excessive weld metal deposited, excessive undercut, porosity, slag, etc.

Cause of Deficiencies —

In the case of work performed by Bechtel, the cause of the deficiencies includes failure of inspectors to obtain and use suitable weld gauges and insufficient design information available to properly apply inspection criteria. In the case of work performed by UMEC, it includes failure to perform 100% inspection and use of a sampling approach instead.

Analysis of Safety Implications

An extensive review has established that the specified criteria for this welding are highly conservative with respect to the requirements of the design. Accordingly, in most cases the deficiencies do not imply a lack of sufficient safety. In some cases, however, repairs have been made in order to assure desired margins of safety. In these areas, deficiencies in structural steel and miscellaneous metal might have resulted in a joint failure under design conditions.

Corrective Action

Review of Welding Requirements —

As noted above, review of specified welding requirements disclosed that the criteria of AWS D1.1-1972 exceed those necessary and appropriate for many applications. These criteria are directed principally toward the fabrication of heavy rolled shapes or built-up sections and are not specifically intended

for application to lighter gauge miscellaneous metal fabrication. Also, the structures, systems and components under review are basically designed for static loading or low cycle fatigue conditions and consequently the potential for brittle failure due to fatigue is not of significant consideration. Thus, the importance of geometric discontinuities within a weld profile influence the structural integrity of the weld only to the extent that they reduce the member thickness or the load resisting capability of the joint configuration.

In accordance with provisions in the Code for the Engineer to specify criteria to comply with the design intent of the Engineer, a classification of weld joints has been established as shown in Appendix A. Criteria are provided in Appendix A for each joint classification. These criteria will be used by Bechtel during inspection of future structural steel and miscellaneous metal welding. Similar considerations apply to work by UMEC.

Use of the criteria of Appendix A in lieu of uniform application of the criteria of AWS D1.1-1972 has been implemented in accordance with requirements of our design control procedures and will be reflected in the FSAR.

Reinspection of Prior Work —

All accessible Bechtel work is being reinspected, and repaired where necessary, to meet specified welding requirements. Paint and fireproofing are being removed as required to permit this reinspection. Similarly, accessible UMEC work not previously inspected is being inspected and repaired as necessary to meet requirements. Only a small amount of work in each case will not be accessible for this reinspection.

The status of the Bechtel reinspection program is shown in Appendix B. The status of the UMEC reinspection program is as follows.

Number of welds to be inspected	12,000
Number inspected	10,000
Number inaccessible for inspection	300
Number inspected requiring repair	270

2.12 A00L

Both reinspection programs are due to be completed by June 30, 1977.

Action to Prevent Recurrence —

In the case of work performed by Bechtel, weld gauges have been obtained for use in visual weld inspection, training sessions have been held for inspectors to instruct them in the proper use of these gauges and additional design information has been provided to clarify welding acceptance standards. In the case of work performed by UMEC, the following action has been taken.

- The appropriate installation inspection supplement sheets were provided with a hold point for weld inspection by UMEC quality control.
- Training sessions were held for UMEC personnel involved to instruct them on inspection requirements.
- Additional inspection personnel were added to UMEC quality control.
- Bechtel increased the level of surveillance inspection of UMEC welding inspection activities.

CONCLUSION

The corrective action described above will provide assurance that structural steel and miscellaneous metal welding for San Onofre Units 2 and 3 conforms with requirements appropriate to the design application and thus that the weld joint will perform its intended safety function.

VISUAL INSPECTION CRITERIA FOR STRUCTURAL STEEL AND MISCELLANEOUS METAL WELDING TO MEET DESIGN REQUIREMENTS

1.0 SCOPE

This Specification provides the acceptance criteria for visual inspection of the welding of structural steel and miscellaneous metal performed in accordance with Specifications referencing AWS D1.1-72. This criteria reflects design requirements consistent with engineering approval specified in AWS D1.1-72, Sections 1.1.2, 3.1.4, 3.7.4, and 3.7.5. It also includes welding of light gage material in HVAC ductwork whose classification is not specifically covered by AWS D1.1-72.

2.0 CLASSIFICATION OF WELD JOINTS

The following classification of weld joints will satisfy the design intent. In addition this classification of weld joints shall be based upon suitability for service in accordance with the following categories.

2.1 Category A Joints are part of the main building frame and carry principal design loads.

2.2 Category B Joints are connections between main building frame and miscellaneous metal.

2.3 Category C Joints are not part of the main building frame, but rather provide auxiliary support or framing for systems, components and equipment. These joints are within the miscellaneous metal category, and shall include, but are not limited to, pipe supports beyond the scope of ASME, stairways, embedments, instrument supports, electrical raceway and supports, and HVAC duct supports.

2.4 Category D Joints are not part of the building frame, or auxiliary support system but rather perform a passive or inactive function. These joints are within the miscellaneous metal category and shall include, but are not limited to, doors, windows, hatch covers and frames, ledger angles, hand rails and gratings.

2.5 Category E Joints are limited to those welds, used in ductwork welding of thin walled gage steel, whose classification is not specifically covered by AWS D1.1-72.

3.0 ACCEPTANCE CRITERIA

The basis for acceptance shall be based on the type of joint and suitability for purpose.

3.1 Category A Joints shall be acceptable provided the following requirements are met.

3.1.1 The weld meets or exceed specified size requirements. Either or both fillet weld legs may exceed design size by 1/8 inch for welds up to and including 5/16 inch fillet, and 1/4 inch for welds larger than 5/16 inch fillet. Welds may be longer than specified. Continuous welds may be accepted in place of intermittent welds.

3.1.2 The fillet leg dimension may not under run the nominal fillet size by more than 1/16 inch for more than 10 percent of the weld length. For flange to web joints the under size may not be within two flange thicknesses of the weld end.

3.1.3 The weld may contain a maximum of five percent by surface area unaligned, unclustered porosity.

3.1.4 Convexity height and roll over may not exceed 1/8 inch. Butt weld reinforcement may not exceed 1/8 inch.

3.1.5 The weld may have an underfilled crater, provided the underfill depth does not exceed 1/32 inch, and the crater has a smooth contour blending gradually with the adjacent weld and base metal without acute notches.

3.1.6 Undercut not exceeding 1/32 inch deep may be accepted when oriented parallel to the longitudinal axis of the base metal and for joints which traverse 50 percent or less of base metal width. Weld lengths which exceed 50 percent base metal

width may have undercut not exceeding 0.01 inch deep. In case of uncertainty, reference shall be made to the project workmanship acceptance standard for buildings. These standards represent physical welding samples which have been reviewed and approved by engineering and represent workmanship standards established for specific use by field inspection personnel. Work equal to or better than this standard shall be accepted.

3.1.7 There are no cracks in the weld.

3.1.8 Thorough fusion exists between weld metal and base metal, except as permitted in 3.1.4.

3.2 Category B Joints shall be acceptable based upon the same criteria as applied to Category A Joints.

3.3 Category C Joints shall be acceptable provided the following requirements are met.

3.3.1 All conditions described as acceptable in Paragraphs 3.1.1, 3.1.2, 3.1.3, 3.1.4, 3.1.7 and 3.1.8 are acceptable for Type C Joints.

3.3.2 Undercut (underfill) not exceeding 1/32 inch may be acceptable for the full length of the weld. Undercut not exceeding 1/16 inch may be accepted provided it is wider than deep and does not have an acute intersection at its root. The accumulative length of 1/16 inch undercut shall not exceed 50% of the weld length. For members welded from both sides the cumulative undercut depth or length shall not exceed the criteria. In case of uncertainty, undercut equal to or less severe than shown in the project workmanship acceptance standards for non-building applications, shall be accepted.

3.3.3 Underfilled groove weld craters shall be accepted provided the depth of underfill is 1/16 inch or less. Underfilled single-pass fillet weld craters shall be accepted provided the crater length is less than ten percent of the weld length. On multi-pass fillet welds crater depth 1/16 inch or less shall be accepted.

3.4 Category D Joints shall be acceptable provided the following requirements are met.

3.4.1 All of the conditions described as acceptable in paragraphs 3.1.1, 3.1.2, 3.1.4, 3.1.7 and 3.1.8.

3.4.2 All of the conditions described as acceptable in paragraphs 3.3.3.

3.4.3 Porosity or slag inclusions are not a criteria for rejection.

3.4.4 Undercut shall not exceed 50 percent of the material thickness.

3.5 Category E Joints shall be acceptable provided the following requirements are met.

3.5.1 All conditions described as acceptable in paragraphs 3.1.1, 3.1.2, 3.1.3, 3.1.4, 3.1.7, 3.1.8, and 3.4.4 are acceptable for Category E Joints.

3.5.2 Faying surfaces shall not exceed 3/16 inch gap between the parts to be joined. The leg of the fillet weld size will be increased by the amount of the separation.

3.5.3 Abutting parts to be joined by butt welds shall be carefully aligned and misalignment shall not exceed the thickness of the thinner material being welded as measured from the highest abutting member. In no case will more than 1/8 inch misalignment be permitted as a departure from the theoretical alignment.

3.5.4 Butt weld profile shall be convex and reinforcement or crowning shall not exceed 1/8 inches. The welded joint for ductwork shall develop complete penetration a minimum of 80% of the length of the welded joint.

3.5.5 Corner welds used to seal ductwork are designated partial penetration welds. Such welds do not require full fusion and weld reinforcement greater than the material thickness shall verify the adequacy of the weld, provided that the toes of the weld have complete fusion.

3.5.6 Face of fillet welds may be slightly convex, flat or slightly concave. Convexity shall not exceed 1/8 inches. Concavity shall not reduce the weld throat beyond that required for weld size. Fillet weld size shall be as required by structural calculations, with minimum sizes as indicated in AWS D1.1-72 Section 2.7.1.1.

3.5.7 Turning vanes and turning vane rails which are fabricated of light gauge material and welded to heavier gauge ductwork will be welded with a fillet weld as required by design drawings. These fillet welds may exceed the profile and convexity limits as previously described and are acceptable for this application. Minor burn-through cannot be avoided on vanes, and will be permitted up to 1/4 inch in length provided equivalent length of fillet welds are added to compensate for welds weakened by burn-through.

3.5.8 Burn through on partial penetration welds is permitted provided leak-tight integrity is maintained. Metal flow on the inside of the duct is permitted provided it is fused completely with the parent metal and metal thickness is not reduced by greater than 50%.

3.5.9 Scratching of metal in fit-up and isolated arc strikes must be removed only to the extent necessary to remove sharp burrs. The intent of this stipulation is to limit excess grinding of base metal, provided it does not exceed 50% of the base metal thickness in these isolated areas.

3.5.10 Distortion caused by welding longitudinal seams shall not exceed 2% of the nominal diameter measured from the cross-sectional cord of the distorted area.

FOR BECHTEL FIELD WELDING

Area/Craft	Number of Welds	Number Inspected	Number Requiring Repair	Number Accepted to Date
Control				
Electrical	13000	8500	3900	8110
P.S.	85	0	0	0
Structural	200	176 (1)	(*)	176
HVAC	1600	1600	160	1600
Radwaste				
Electrical	519	519	51	519
P.S.	175	17	2	15
Structural	24	24	0	24
HVAC	642	642	106	594
Containment #2				
Electrical	78	78	(*)	78
P.S.	0	0	0	0
Structural	240	160	(*)	160
HVAC	0	0	0	0
Penetration #2				
Electrical	200	200	75	200
P.S.	10	4	3	4
Structural	180	172 (2)	(*)	172
HVAC	0	0	0	0
Safety Equipment Bldg. #2				
Electrical	450	350	100	350
P.S.	30	22	0	22
Structural	8	0	0	0
HVAC	170	170	0	0
	17611	12470	-	12020

Notes:

(1) Field welding of 24 column stubs at the roof elevation of the Control Area are embedded in concrete and consequently unaccessible for reinspection.

(2) Field welding of beam to column connections in the Penetration Area are embedded in concrete and consequently unaccessible for reinspection.

(*) Data not available.

JUN 0 6 1977

Distribution
Docket File
LWR #2 File
MMMlynczak
HRood
KKniel
DBVassallo
RCDeYoung

Docket Nos. 50-361
and 50-362

MEMORANDUM FOR: Edson G. Case, Acting Director, Office of Nuclear Reactor Regulation

FROM: Roger S. Boyd, Director, Division of Project Management, NRR

SUBJECT: LEVEL D SCHEDULE FOR SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 OPERATING LICENSE REVIEW

Enclosed for your review and approval are the Level C summary of review milestones and the Level D schedule for the operating license safety review of the San Onofre Nuclear Generating Station, Units 2 and 3.

The Final Safety Analysis Report was docketed on March 23, 1977. The Safety Evaluation Report will be issued April 21, 1978. The prospective ASLB decision date is January 19, 1979, which is prior to the applicant's targeted fuel loading dates of February 1980 (Unit 2) and May 1981 (Unit 3). Thus, this schedule satisfies the objective of completing licensing action prior to the anticipated fuel loading dates.

Original signed by:
Roger S. Boyd

R. S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosures:
1. Level C Summary
2. Level D Schedule

My

OFFICE	DPM:LWR #2	DPM:LWR #2	DPM:LWR #2	DPM:AD/LWR	DD:DPM	DPM
SURNAME	MMMlynczak:mt	HRood	KKniel	DBVassallo	RCDeYoung	RSBoyd
DATE	6/ /77	6/ /77	6/ /77	6/ /77	6/ /77	6/ /77

ENCLOSURE 1

LEVEL C MILESTONE SUMMARY

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

Milestone	Description	Elapsed Weeks	Date	Reactor Systems Bran[ch] Date
1-1	FSAR Tendered	-	11/30/76	
1-6	FSAR Docketed	0	3/23/77	
5-0	Q-1's to LPM	10	6/3/77	8/15/77
8-0	Q-1's to Applicant	13	6/24/77	
9-0	Applicant Response to Q-1's	21	8/19/77	
12-0	Formal Staff Positions to LPM	29	10/14/77	12/15/77
14-0	Formal Staff Positions to Applicant	32	11/4/77	
15-0	Applicant Response to Staff Positions	40	12/30/77	
24-0	SE Input to LPM	48	2/24/78	
24-90	Draft SER by LPM	52	3/24/78	
24-98	Management Review SER Complete	54	4/7/78	
25-0	SER Issued	56	4/21/78	
25-8	Full ACRS Meeting	63	6/9/78	
27-0	ACRS Letter Received	64	6/16/78	
28-0	SER Supplement Issued	72	8/11/78	
38-95	Start of Rad/Safety Hearing	-	9/12/78	
39-71	End of Rad/Safety Hearing	-	11/10/78	
39-90	ASLB Decision	-	1/19/79	

Milestone	Description	Elapsed Weeks	Date	Reactor Systems Brar Date
40-0	Licensing Effort Complete	-	1/19/79	
42-0	Fuel Loading - Unit 2	-	2/80	
	Fuel Loading - Unit 3	-	5/81	

Docket Files

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

JUN 0 3 1977

Docket Nos. 50-361
50-362

MEMORANDUM FOR: K. Kniel, Chief, Light Water Reactors Branch No. 2, DPM

FROM: V. Benaroya, Chief, Auxiliary Systems Branch, DSS

SUBJECT: FIRST ROUND REQUEST FOR ADDITIONAL INFORMATION - SAN ONOFRE NUCLEAR GENERATING STATION, UNITS NO. 2 & 3

Plant Name:	San Onofre Nuclear Generating Station, Units No. 2 & 3
Licensing Stage:	OL
Docket Numbers:	50-361/362
Milestone Number:	05-02
Responsible Branch and Project Manager:	LWR 2 H. Rood
Requested Completion Date:	June 3, 1977
Review Status:	Awaiting Information

The enclosed first round request for additional information covers those portions of the San Onofre Nuclear Generating Station, Units No. 2 and No. 3 FSAR for which the Auxiliary Systems Branch has primary responsibility. Our review is based on the San Onofre Nuclear Generating Station, Units No. 2 and No. 3 FSAR up to and including Amendment 1.

The enclosure identifies areas where we need additional information. These areas include flood protection, protection against postulated rupture of piping, new and spent fuel storage, fuel pool cooling system, fuel handling system, salt water system, component cooling water system, ultimate heat sink, condensate storage facilities, auxiliary building ventilation systems, support building ventilating systems, main steam system upstream of the main steam isolation valve, circulating water system and auxiliary feedwater system.

Contact:
C. Liang
Ext. 27763

MY

In Amendment 1 of the FSAR, the applicant states that they will submit the results of fire protection re-evaluation and high energy line failure analyses in October, 1977 and August, 1977, respectively. We will prepare our request for additional information regarding these two subjects after receipt of the applicant's submittals.

Donald C Fischer /for

Victor Benaroya, Chief
Auxiliary Systems Branch
Division of Systems Safety

Enclosure:
As stated

cc: S. Hanauer
R. Heineman
R. Boyd
D. Vassallo
W. McDonald
R. Tedesco
H. Rood
V. Benaroya
D. Fischer
P. Matthews
J. Glynn
C. Liang
FILE: San Onofre 2 & 3

010.0 AUXILIARY SYSTEMS BRANCH

010.20 (3.4)

You state in Section 4.3.1.2 of the FSAR that rooms containing non-seismic Category I system components and pipes whose rupture could result in flood damage to equipment important to safety have level alarms that alarm in the control room. Provide design classifications of the level monitoring and alarm systems and their capabilities of meeting the single failure criterion. Also describe the action that will be taken to prevent safety related equipment from being flooded, taking into account time for operator manual action; namely, 20 minutes manual action time should be assumed if only a single action is required inside the control room or 30 minutes manual action time should be assumed if there is more than one operator action required inside the control room.

010.21 (RSP) (3.6)

Your response to our request 010.4 is not acceptable. It is our position that San Onofre Units No. 2 and No. 3 follow the guidance provided in the December, 1972 letter from A. Giambusso and the corrections in the errata sheet for "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment" that was transmitted to Mr. J. Moore and Mr. Engler from Mr. K. Goller dated January 22, 1973. Also, an analyses made in conformance with B.3 of our Branch Technical Position APCSB 3-1 must be presented to demonstrate that acceptable protection against the effects of piping failures outside containment has been provided.

010.22 (9.0)

Provide a tabulation of all valves in the reactor pressure boundary and in other seismic Category I systems (per Regulatory Guide 1.29) e.g., safety valves, relief valves, stop valves, stop-check valves, and control valves whose operation is relied upon either to assure safe plant shutdown or to mitigate the consequences of a transient or accident. The tabulation should identify the system in which it is installed, the type and size of valves, the actuation type(s), and the environment of conditions to which the valves are qualified.

010.23 (RSP) (9.1.1)

It is our position that the spacing between fuel assemblies in the new fuel storage racks should be sufficient to maintain the array, when fully loaded and flooded with non-borated water, in a subcritical condition, i.e., K_{eff} of less than 0.95. Provide additional information in your FSAR to demonstrate that the above criteria are met.

010.24 (9.1.1) (9.1.2) Provide results of an analysis to demonstrate that the new and spent fuel storage racks and the anchorages can withstand the maximum uplift forces available from the crane without an increase in K_{eff} and verify that the vaults and racks have been designed to preclude damage from dropped heavy objects.

010.25 (9.1.3) Section 9.1.3 of the FSAR states that the makeup to the spent fuel pool is from the seismic Category I refueling water storage tank. Show this makeup line in your Figure 9.1-1 accordingly.

010.26 (9.1.4) In addition to the fuel handling arrangement in Figure 9.1-16, provide additional drawings including section views that show more clearly the relationship between the spent fuel pool and the spent fuel cask area. These drawings should be sufficiently detailed to demonstrate that the spent fuel cask cannot be hoisted to a height that would permit possible tipping or swinging of the cask over the spent fuel storage pool during cask handling.

010.27 (3.2.1) (9.1.4) Table 3.2-1 of the FSAR indicates that the refueling machine, spent fuel handling machine, new fuel crane and new fuel elevator are not designed to seismic Category I standards. Expand Section 9.1.4.3 of the FSAR to discuss the consequences of a failure due to a postulated SSE of the above fuel handling equipment to essential fuel handling and storage facilities in accordance with Position C.2 of Regulatory Guide 1.29.

010.28 (9.2.1) Table 9.2-1 of the FSAR indicates that the salt water cooling system required flow for LOCA conditions is 34,000 gpm. Section 9.2.1.2 states that each pump with a capacity of 17,000 gpm is capable of providing 100 percent of the cooling flow required to mitigate the effects of any design basis accident including a LOCA. Provide clarification for this discrepancy.

010.29 (RSP) (9.2.2) Your response to our request 010.13 is not acceptable. It is our position that the San Onofre Units No. 2 and No. 3 should modify the design of the component cooling water system supplying cooling water to the reactor coolant pumps to follow the guidance set forth in our request 010.13. Your response also stated that you do not consider moderate energy line cracks to be design criteria for San Onofre Units No. 2 and No. 3. Refer to our request 010.21. Failures in moderate energy component cooling water piping systems should be considered for San Onofre Units No. 2 and No. 3 design.

010.30 (9.2.2) Section 9.2.2 of your FSAR indicates that the reactor coolant pumps and the spent fuel pool heat exchangers are supplied component cooling water (CCW) from a non-essential loop of the CCW system which will be isolated under accident conditions. Describe the provisions made to reopen the non-critical loop isolation valves so that the CCW can be supplied to the fuel pool heat exchangers before boiling in the spent fuel storage pool occurs.

010.31 (9.2.2) Describe and provide the design bases for the makeup water source to the surge tank of the component water system. Include the seismic category classification of the makeup system.

010.32 (9.2.5) Your response to our request 010.11 indicates that the traveling water screen for the salt water system is not designed to seismic Category I requirements. Also Section 9.2.5 of the FSAR states that the offshore intake and outfall conduit structures are not designed to seismic Category I requirements. Provide results of an analysis to demonstrate that the salt water cooling pumps will be provided with required suction flow under a postulated LOCA assuming the effects of failure of non-seismic designed components and loss of offsite power conditions. The analysis should also include the effects of a postulated failure of non-seismic designed seawall which is located above the intake and outfall conduits.

010.33 (9.2.6) Section 9.2.6.3 of your FSAR indicates that the condensate storage capacity for the auxiliary feedwater supply is based on a cooldown time of about 3 1/2 hours. Your response to our request 010.14 refers to a cooldown time of 4.2 hours assumed in the accident analysis. Clarify this deviation to assure that the condensate storage capacity is sufficient for plant cold shutdown.

010.34 (9.3.4) Provide page 9.3-36 of the FSAR for our review. This page is now missing from the FSAR.

010.35 (9.4.2) Figure 9.4-8 of the FSAR indicates various points of the control room complex emergency HVAC system where the air enters or leaves the system on the same drawing. Provide additional information to clearly identify the points that should be interconnected on continuation drawings (i.e. match points).

010.36 (9.4.2) Figure 9.4-8 of the FSAR indicates that there are no isolation dampers between the emergency air cooling system and the normal air cooling system for the ESF switchgear rooms. Revise the P&ID and the description of the ESF switchgear room cooling system to demonstrate that provisions are made to isolate the normal cooling system when the emergency cooling system is in operation.

010.37 (9.4.2) Section 9.4.2 of the FSAR indicates that two emergency exhaust fans serve all four battery rooms for each unit. Each emergency fan connects to two battery rooms. A single failure on one emergency exhaust fan will result in loss of exhaust capability of two battery rooms. This is not acceptable. Since hydrogen may buildup in these battery rooms. Modify the system to provide continuous exhaust for all battery rooms under loss of offsite power conditions and meet the single failure criterion.

010.38 (9.4.3) In Section 9.4.3.1 of the FSAR you state that the fuel handling building (FHB) normal ventilation system isolation dampers are pneumatically operated. Figure 9.4-9 indicates that these isolation dampers are motor operated. Clarify this discrepancy in your FSAR. If they are air operated, describe the safety classification of the supply air and the failure mode of the isolation dampers in case of loss of air supply. If they are motor operated, describe the electric power and instrumentation supply to the isolation dampers and demonstrate that a single electric failure assuming loss of offsite power will not prevent positive isolation of the FHB normal ventilation system.

010.39 (9.4.3) Section 9.4.3.1 of the FSAR states that the post-accident cleanup units and fuel pool pump room cooling units are powered from a vital bus. Clarify this statement to confirm that the redundant cleanup units and redundant cooling units are powered from separate vital buses to meet the single failure criterion.

010.40 (9.4.3) You stated that the description of the design of the diesel generator building ventilation system would be submitted in April, 1977. This information has not be received. This information is necessary for us to evaluate this system.

010.41 (10.3) Identify on Figure 10.1-1 of the FSAR, the seismic Category I portion of the control air system for operation of the power operated atmospheric steam dump valves and identify the interface between the seismic Category I nitrogen supply line and the non-seismic Category I instrument air supply line.

010.42 (RSP) (10.4.5) In your response to our request 010.17, you have assumed an operator reaction time of 5 minutes and 38 seconds from the first flood alarm to trip the circulating water pump motor in a failed line. This is not acceptable. It is our position that 20 minutes manual action time should be assumed (from the time of break) if only a single action is required inside the control room to trip the circulating water pump motor or 30 minutes manual action time should be assumed if there is more than one operator action required inside the control room. Re-evaluate the postulated circulating water system failure based on the above stated manual action times.

010.43 (10.4.9) Provide additional detailed P&IDs of the auxiliary feedwater systems to indicate design features such as the following:

(1) The alternative water supplies to the auxiliary feedwater pumps.

(2) Show the steam supply lines from the main steam lines upstream of the MSIVs to the turbine driven auxiliary feedwater pump turbine and indicate that the motor operated valve on the steam supply line is powered from D/C power sources and the air operated valves are operated by safety grade air supplies.

010.44 (10.4.9) Provide the results of an analysis to demonstrate that your auxiliary feedwater pump size and condensate storage capacity for the auxiliary feedwater supply is sufficient to prevent overheating and subsequent overpressurization of the primary coolant system under all postulated accident conditions including the following situations:

(1) Main steam or feedwater line failure and both auxiliary feedwater pumps start to pump water through the break until the operator manually isolates the steam generator with the broken line. It is our position that 20 minutes manual action time should be assumed if only a single action is required inside the control room to isolate the line break or 30 minutes manual action time should be assumed if more than one operator action is required inside the control room.

(2) A moderate energy line crack at the condensate supply line close to the tank T 121.

Docket File 50-362

JUN 0 3 1977

Docket Nos. STN 50-361/362

MEMORANDUM FOR: K. Kniel, Chief, Light Water Reactors Branch No. 2, DPM

FROM: J. Collins, Chief, Effluent Treatment Systems Branch, DSE

SUBJECT: ROUND ONE QUESTIONS FOR SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NOS. 2 AND 3

PLANT NAME: San Onofre Nuclear Generating Station
LICENSING STAGE: OL
DOCKET NUMBERS: 50-361/362
MILESTONE NUMBER: 5-3
RESPONSIBLE BRANCH: LWR No. 2
PROJECT MANAGER: H. Rood
DESCRIPTION OF RESPONSE: Round One Questions
REQUESTED COMPLETION DATE: June 3, 1977
REVIEW STATUS: Additional information needed

We have reviewed the PSAR for San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, and find that we need additional information to complete our evaluation. The applicant has not provided sufficient information concerning the ESF atmosphere cleanup systems, instrumentation for the steam generator blowdown system, tanks outside containment containing potentially radioactive materials, automatic gas analyzers for potentially explosive systems, and the bases for your cost-benefit analysis performed for Appendix I of 10 CFR Part 50.

We will need the additional information by August 19, 1977, to meet our schedule.

ORIGINAL SIGNED BY
J. T. COLLINS

John T. Collins, Chief
Effluent Treatment Systems Branch
Division of Site Safety and
Environmental Analysis

Enclosure:
Round One Questions

cc: See Page 2

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DATE	06/03/77	06/ /77	06/ /77			

cc: S. Hanauer
H. Denton
D. Muller
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R. Boyd
D. Vassallo
H. Rood
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W. Kreger
W. McDonald (w/o encl)
W. Burke
P. Stoddart

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DRAFT Q-1's FOR SAN ONOFRE
UNIT NOS. 2 & 3
FSAR REVIEW

321.1 (6.5.1; 9.4.3.1) Your description of the engineered safety feature filter systems is not adequate. Provide your justification for not including demisters in the engineered safety features filter system for the fuel handling building.

321.2 (10.4.8 SRP) Describe your provisions for instrumentation and control of the steam generator blowdown system to protect temperature sensitive elements, such as demineralizers, and to control flushing, liquid levels, and process flows through individual components.

321.3 (11.2) For gaseous and liquid radioactive waste processing systems, provide in tabular form a comparison between the components of your proposed systems and the appropriate equipment codes presented in Table 1 of Branch Technical Position ETSB 11-1 (Rev. 1), a copy of which is attached. Your tabulation referenced in Chapter 3.2 of the FSAR is not satisfactory in that it is not complete with respect to Materials, Welder Qualification and Procedure, and Inspection and Testing.

321.4 (11.2) Provide a table listing tanks outside reactor containment which contain potentially radioactive liquid materials. The table should include tanks located both inside and outside of plant buildings. For each tank, indicate the provisions incorporated to monitor tank levels, to annunciate potential overflow conditions, and to collect and process liquids in the event of overflows. Acceptable provisions include dikes around tanks, retention basins, and elevated thresholds to contain liquids in bays containing the tanks.

321.5 (11.3; 9.3.2) Your response to question 320.2 is not satisfactory. It is our position that at least one additional gas analyzer, which is continuously on stream at a point common to streams measured sequentially, should be added to your system. It is also our position that the gas analyzers should, upon high-high alarm, initiate automatic control features to reduce the potential for explosion; acceptable automatic control features which should be considered are automatic isolation of either the source of oxygen or hydrogen or the injection of diluents to reduce concentrations to limits outside of the explosive envelope.

321.6 (11.3.1.7) Your cost-benefit analysis, as provided in Appendix 5A of the Environmental Report and referenced in Section 11.3, does not present sufficient information to permit us to evaluate your results. You should provide the bases for your cost-benefit analysis in the form of Cost Estimate Sheets, as shown in Appendix B of Regulatory Guide 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors," March 1976, for each augment considered in your analysis.

321.7 (10.4.3) Section 10.4.3.2.2, seventh paragraph, states: "A full discussion of the radiological aspects of primary-to-secondary system leakage, including anticipated releases from the turbine gland sealing system...is included in Chapter 11." We did not find the referenced discussion in Chapter 11. You should provide the referenced discussion.

321.8 (11.4) Your waste solidification system provides for mixing paddles to be used only when solidifying resins. Describe your procedures for verifying that adequate mixing can be achieved when solidifying materials other than resins without mechanical mixing in the solidification container and for assuring the absence of free water in the completed product for all types of materials to be solidified.

Dockets

JUN 3 1977

Docket Nos. 50-361
and 50-362

MEMORANDUM FOR: William H. Regan, Jr., Chief
Environmental Projects Branch 2, DSE

FROM: Robert B. Samworth, Section Leader
Aquatic Resources Section
Environmental Specialists Branch, DSE

SUBJECT: STAFF INPUT TO EPM - FORMAL QUESTIONS FOR SAN ONOFRE - UNITS 2 & 3 OL REVIEW

PLANT NAME: San Onofre Nuclear Generating Station, Units 2 & 3
LICENSING STAGE: OL
DOCKET NUMBER: 50-361 and 50-362
RESPONSIBLE BRANCH: Environmental Projects Branch 2
PROJECT MANAGER: O. D. T. Lynch, Jr.
REQUESTED COMPLETION DATE: June 6, 1977
DESCRIPTION OF RESPONSE: Staff comments on Draft Formal Questions to Applicant
REVIEW STATUS: Aquatic Resources Section input complete

The Aquatic Resources Section has reviewed Draft Formal Questions as requested by note from the EPM on May 26, 1977. We have determined that Draft Question 1.2 has been adequately addressed in the applicant's Amendment 1 to the ER (OL Stage), page 5.1-8, hence should be deleted. All other draft comments are satisfactory and should be transmitted as formal questions.

The following "new" question, developed as a result of the site visit, should be added to the formal Request for Additional Information:

M4

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"Provide design and operational characteristics for the Fish Conservation System, including size of fish basket, tolerance between basket and walls, estimated frequency of use, and methodology for estimating entrapment rates and survival rates for fishes returned via the system."

This review was performed by R. Romano and C. Billups.

Original Signed by Robert B. Samworth

Robert B. Samworth, Section Leader
Aquatic Resources Section
Environmental Specialists Branch
Division of Site Safety and
Environmental Analysis

cc: H. Denton
D. Muller
M. Ernst
V. Moore
O. Lynch
R. Ballard
R. Samworth
R. Romano
C. Billups

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DATE	6/3/77	6/3/77	6/3/77			

Docket 50-362

JUN 1 1977

Docket Nos.: 50-361
50-362

MEMORANDUM FOR: K. Kniel, Chief, Light Water Reactors Branch #2
Division of Project Management

FROM: C. J. Heltemes, Jr., Chief, Quality Assurance
Branch, Division of Project Management

SUBJECT: HELTEMES/KNIEL - MAY 27, 1977 MEMO REQUESTING
ADDITIONAL INFORMATION FOR QA REVIEW OF SAN ONOFRE
UNIT NOS. 2 & 3

Question 421.1 in the subject memo should be changed in total to read as follows:

> "SCE is requested to describe in the FSAR their QA program for fire protection in accordance with the information and guidance previously transmitted with the Boyd/Moore letters dated May 3, 1976 and September 30, 1976."

Original signed by:
J. W. Gilray

C. J. Heltemes, Jr., Chief
Quality Assurance Branch
Division of Project Management

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SURNAME ➤	JTConway:jk	JWGilray	CJHeltemes			
DATE ➤	6/1/77	6/1/77	6/1/77			

NRC FORM 318 (9-76) NRCM 0240

Docket 50-362

MAY 27 1977

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HBClayton
AMGarland

MEMORANDUM FOR: K. Kniel, Chief, Light Water RNactors Branch #2, DPM

FROM: C. J. Heltemes, Jr., Chief, Quality Assurance Branch, DPM

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION: SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NOS. 2 and 3: QUALITY ASSURANCE, CONDUCT OF OPERATIONS, INITIAL TEST PROGRAM

Applicant: Southern California Edison Company (SCE)
Licensing Stage: OL
Responsible Branch: LWR#2
Project Manager: H. Rood
Requested Completion Date: June 3, 1977
Review Status: Q-1's Complete

The Quality Assurance Branch has reviewed the FSAR sections describing the Conduct of Operations (Sections 13.1 and 13.4), Initial Test Program (Chapter 14), and the Quality Assurance Program (Section 17.2) of the FSAR submitted in support of the Operating License application for San Onofre, Unit Nos. 2 and 3.

The QA program for operations is described in Section 17.2 of the QA topical report, SCE-1, Amendment 1, dated March 1977. This report is currently being reviewed by the QAB, and the estimated completion date is June 17, 1977.

Additional information is required from SCE on Sections 13.1, 13.4, and 17.2 and Chapter 14. Accordingly, SCE should be requested to amend the appropriate section and chapter of the FSAR to satisfy the enclosed request prior to August 19, 1977, in order for the QAB to maintain their review schedule.

We are available for a meeting or conference call with SCE to discuss these questions.

Original signed by:
R. J. McDermott

C. J. Heltemes, Jr., Chief
Quality Assurance Branch
Division of Project Management

Enclosure
Request for Additional Information

OFFICE	DPM:QAB	DPM:QAB	DPM:QAB	DPM:QAB	DPM:QAB
SURNAME	JTConway:jk	FAllenspach	HBClayton	JWGilray	CJHeltemes
DATE	5/ /77	5/ /77	5/27/77	5/26/77	5/27/77

421.0 QUALITY ASSURANCE

421.1
(17.2)

The QA program should include provisions for fire protection and must be described in the FSAR. Page 9.5.1-37 of the May 1, 1976 revision of Section 9.5.1 of NUREG 75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition" provides the NRC position relative to the QA program for fire protection. Our acceptance criteria, which have not yet received final NRC approval, are attached and forwarded for your guidance.

Quality Assurance Program Acceptance Criteria

The quality assurance (QA) program should be capable of verifying that the requirements for design, procurement, installation, and testing and administrative controls for the fire protection program for safety-related equipment as defined in Regulatory Guide 1.120 are satisfied. In the case of operating plants, quality assurance should be established to verify that the requirements of the fire protection program approved by NRC are satisfied. The QA program should be under the management control of the QA organization. These QA criteria apply to those items within the scope of the fire protection program, such as fire protection systems, emergency lighting, communication and breathing equipment as well as applicable safety related equipment. The description of how to satisfy the criteria specified in the Branch Position is acceptable if it includes the following:

(a) Design Control and Procurement Document Control - Measures should be established to assure that the applicable guidelines of the Regulatory Guide 1.120 or approved NRC alternatives are included in design and procurement documents and that deviations therefrom are controlled. These measures should assure that:

1) Design and procurement document changes, including field changes and design deviations are subject to the same level of controls, reviews and approvals that were applicable to the original document.

2) Quality standards are specified in the design documents such as appropriate fire protection codes and standards, and deviations and changes from these quality standards are controlled.

3) New designs and plant modifications, including fire protection systems, are reviewed by qualified personnel to assure inclusion of appropriate fire protection requirements. These reviews should include items such as:

(i) Design reviews to verify adequacy of wiring isolation.

(ii) Design reviews to verify appropriate requirements for room isolation (sealing penetrations, floors, and other barriers).

4) A review and concurrence of the adequacy of fire protection requirements and quality requirements stated in procurement documents are performed and documented by qualified personnel. This review should determine that fire protection requirements and quality requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and the procurement document has been prepared, reviewed, and approved in accordance with QA program requirements.

(b) Instructions, Procedures, and Drawings - Inspections, tests, administrative controls, fire drills and training that govern the fire protection program should be prescribed by documented instructions, procedures or drawings and should be accomplished in accordance with these documents. The following provisions should be included:

1) Indoctrination and training programs for fire prevention and fire fighting are implemented in accordance with documented procedures.

2) Activities such as design, installation, inspection, test, maintenance, and modification of fire protection systems are prescribed and accomplished in accordance with documented instructions, procedures and drawings.

3) Instructions and procedures for design, installation, inspection, test, maintenance, modification and administrative controls are reviewed to assure the proper inclusion of fire protection requirements, such as precautions, control of ignition sources and combustibles, provisions for backup fire protection if the activity requires disabling a fire protection system, and restriction on material substitution unless specifically permitted by design and confirmed by design review.

4) The installation or application of penetration seals and fire retardant coatings is performed by qualified personnel using qualified procedures.

(c) Control of Purchased Material, Equipment, and Services - Measures shall be established to assure that purchased material, equipment and services conform to the procurement documents. These measures should include:

1) Provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor, inspections at suppliers, or receiving inspections.

2) Source or receiving inspection, as a minimum, for those items whose quality cannot be verified after installation.

(d) Inspection - A program for independent inspection of activities affecting fire protection should be established and executed by, or for, the organization performing the activity to verify conformance

with documented installation drawings and test procedures for accomplishing the activities. This program should include:

(1) Inspections of (1) installation, maintenance and modification of fire protection systems; and (2) emergency lighting and communication equipment to assure conformance to design and installation requirements.

(2) Inspection of penetration seals and fire retardant coating installations to verify the activity is satisfactorily completed.

(3) Inspections of cable routing to verify comformance with design requirements.

(4) Inspections to verify that appropriate requirements for room isolation (sealing penetrations, floors, and other barriers) are accomplished during construction.

(5) Measures to assure that inspection personnel are independent from the individuals performing the activity being inspected and are qualified in the design and installation requirements for fire protection.

(6) Inspection procedures, instructions, and check lists which provide for the following:

   a. Identification of characteristics and activities to be inspected.

   b. Identification of the individuals or groups responsible for performing the inspection operation.

   c. Acceptance and rejection criteria.

   d. A description of the method of inspection.

   e. Recording evidence of completing and verifying a manufacturing, inspection, or test operation.

   f. Recording inspector or data recorder and the results of the inspection operation.

(7) Periodic inspections of fire protection systems, breathing equipment, emergency lighting, and communication equipment to assure the acceptable condition of these items.

(8) Periodic inspection of materials subject to degradation such as fire stops, seals and fire retardant coatings to assure that such items have not been damaged or deteriorated.

(e) Test and Test Control - A test program should be established and implemented to ensure that testing is performed and verified by inspection and audit to demonstrate conformance with design and system readiness requirements. The tests should be performed in accordance with written test procedures; test results should be properly evaluated and acted on. The test program should include the following:

(1) Installation Testing - Following construction, modification, repair or replacement, sufficient testing is performed to demonstrate that fire protection systems, emergency lighting and communication equipment will perform satisfactorily in service and that design criteria are met. Written test procedures for installation tests incorporate the requirements and acceptance limits contained in applicable design documents.

(2) Periodic Testing - The schedules and methods for periodic testing are developed and documented. Fire protection equipment, emergency lighting, and communication equipment are tested periodically to assure that the equipment will properly function and continue to meet the design criteria.

(3) Programs are established for QA/QC to verify testing of fire protection systems and to verify qualifications of test personnel.

(4) Test results are documented, evaluated, and their acceptability determined by a qualified responsible individual or group.

(f) Inspection, Test and Operating Status - Measures should be established to provide for the identification of items that have satisfactorily passed required tests and inspections. These measures should include provisions for:

1) Identification by means of tags, labels or similar temporary markings to indicate completion of required inspections and tests, and operating status.

(g) Nonconforming Items - Measures should be established to control items that do not conform to specified requirements to prevent inadvertent use or installation. There measures should include provisions to assure that:

(1) Nonconforming, inoperative, or malfunctioning fire protection systems, emergency lighting, and communication equipment are appropriately tagged or labelled.

(2) The identification, documentation, segregation, review, disposition, and notification to affected organization of nonconforming materials, parts, components, or services are procedurally controlled.

(3) Documentation identifies the nonconforming item; describes the nonconformance and the disposition of the nonconformance; and includes signature approval of the disposition.

(4) Provisions are established identifying those individuals or groups delegated the responsibility and authority for the disposition and approval of nonconforming items.

(h) Corrective Action - Measures shall be established to ensure that conditions adverse to fire protection such as failures, malfunctions, deficiencies, deviations, defective components, uncontrolled combustible material and nonconformances are promptly identified, reported and corrected. These measures should assure:

(1) Procedures are established for evaluation of conditions adverse to fire protection (such as nonconformances, failures, malfunctions, deficiencies, deviations, and defective material and equipment) to determine the necessary corrective action.

(2) In the case of significant or repetitive conditions adverse to fire protection, including fire incidents, the cause of the conditions is determined and analyzed, and prompt corrective action is taken to preclude recurrence. The cause of the condition and the corrective action taken are promptly reported to cognizant levels of management for review and assessment.

(i) Records - Records should be prepared and maintained to furnish evidence that the criteria enumerated above are being met for activities affecting the fire protection program. The following provisions should be included:

(1) Records are identifiable and retrievable and should demonstrate conformance to fire protection requirements. The records should include results of inspections, tests, reviews and audits; nonconformance and corrective action reports; construction, maintenance and modification records; and certified manufacturers'

(2) Record retention requirements are established.

(j) Audits - Audits should be conducted and documented to verify compliance with the fire protection program, including design and procurement documents, instructions, procedures and drawings, and inspection and test activities. The following provisions should be included:

(1) Audits are periodically performed to verify compliance with the administrative controls and implementation of quality assurance criteria including design and procurement instructions, procedures and drawings and inspection and test activities. These audits are performed by QA personnel in accordance with preestablished written procedures or check lists and conducted by trained personnel not having direct responsibilities in the areas being audited.

(2) Audit results are documented and then reviewed with management having responsibility in the area audited.

(3) Followup action is taken by responsible management to correct the deficiencies revealed by the audit.

(4) Audits are annually performed to provide an overall assessment of conformance to fire protection requirements.

422.0 CONDUCT OF OPERATIONS

422.1 (13.1.1) Provide the number of professional persons assigned to the Mechanical, Civil/Structural, Controls and Electrical, Apparatus and Materials, and Nuclear Engineering Sections of the Engineering and Construction Department.

422.2 (13.1.1) Provide the personal resume of the person assigned to the position of Supervisor of Plant Operation and Maintenance (superior of Superintendent of San Onofre Station).

422.3 (13.1.2.2) Describe the responsibilities and authority of the Operating Foreman and Fuel Handling Foreman shown in Figure 13.1-3.

422.4 (RSP) (13.1.3.1) You state in Section 13.1.3.3 that the Supervisor of Plant Operations should have a "total of five years experience ..." It is our position that the Supervisor of Plant Operations' position is comparable to that of Operations Manager described in Section 4.2.2 of ANSI N18.1-1971 and should have a "minimum of eight years of responsible power plant experience ..." State your intent to conform to this position.

422.5 (13.1.3.1) Your description of the minimum requirements for the position of Chemical and Radiation Protection Engineer are not clear in regard to the requirements for a "graduate in engineering or physical science" and the credit toward the number of years experience being fulfilled by related technical or academic training; or the specific experience requirements in chemistry and in radiation protection. Please clarify this item. Note that our position in regard to the minimum requirements for the individual in charge of radiation protection at the site is described in Revision 1 to Regulatory Guide 1.8, September 1975.

422.6 (RSP) (13.1.3.1) You state that the minimum requirements for the positions of Chemical Radiation Foreman, Instrument Foreman and Maintenance Foreman require a high school education or equivalent, two years of appropriate experience and one or two years of training. It is our position that the minimum requirements for these three positions should be as described in Section 4.3.2 of ANSI N18.1-1971; a high school diploma or equivalent and a minimum of four years experience in the craft or discipline he supervises. State your intent to conform to this position.

422.7 (RSP) (13.1.3.1) It is not clear from your description in Section 13.1.3.1 whether or not technicians and repairmen will meet the requirements specified in ANSI N18.1-1971. It is our position that your technicians meet the minimum requirements described in Section 4.5.2 of ANSI N18.1-1971 and that your repairmen meet the minimum requirements described in Section 4.5.3 of ANSI N18.1-1971. State your intent to conform to these positions.

422.8 (13.1.3.1) Describe your minimum qualification requirements for the position of Fuel Handling Foreman.

423.0 INITIAL TEST PROGRAM

423.1 (14.2)

(1) Regulatory Guide 1.68 states that the objectives of the initial test program are to provide assurance that (a) the plant has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public, (b) the plant procedures have been evaluated and demonstrated, and (c) the operating organization is knowledgeable about the plant and procedures and is prepared to operate the facility in a safe manner. The objectives of your test program as stated in Section 14.2.1 do not include Item 3. Expand this section to include this objective.

(2) Section 14.2.2 does not state how the station engineering staff (nuclear engineers and assistants, plant engineer, supervising nuclear engineer, and engineering aides) will be utilized during the initial test program. Expand this section to include this information.

423.2 (14.2.2)

For the staff to complete its review of the organization and staffing of the test program, the following additional or clarifying information will be required:

(1) The responsibilities of the Test Operations Supervisor and the Technical Supervisor of the SCE startup organization.

(2) The minimum qualifications requirements (educational, experience , and nuclear experience) for the following categories of personnel at the time they are assigned to the task. Your response should address all personnel performing the tasks listed and should not be limited to only SCE personnel (e.g., Test Working Group members and augmenting personnel). Note that ANSI N45.2.6, although applicable to some categories of personnel during the construction, preoperational, and startup phases, was not intended to cover personnel in the listed categories.

(a) Personnel that prepare individual preoperational test procedures.

(b) Personnel that supervise or direct the conduct of individual preoperational tests.

(c) Personnel that review and/or approve preoperational test procedures.

(d) Personnel that approve preoperational test results.

(e) Personnel that prepare individual startup test procedures.

(f) Personnel that supervise or direct the conduct of individual startup tests.

(g) Personnel that review and/or approve startup test procedures.

(h) Personnel that approve startup test results.

423.3 (14.2.3) Section 14.2.3 implies that the acceptance review of test procedures will be performed by personnel other than the individual Test Working Group members. Either modify this section to require minimum qualifications for these reviewers that are commensurate with this responsibility or clarify the information presented to indicate that a technical review of test procedures will be performed by the individuals who are members of the TWG.

423.4 (14.2.3) Expand Section 14.2.3 to state which staff positions (functional titles) will write test procedures and review these procedures before they are submitted to the TWG.

423.5 (14.2.4) Section 14.2.4 describes the methods for changing test procedures after they have been approved by the TWG. Define scope/intent and nonscope/intent changes as used in this section.

423.6 (14.2.7) Revise Section 14.2.7 to state where in the test program that the operability of the safety injection tank discharge isolation valves is demonstrated using the emergency power source.

423.7 (14.2.11) For the staff to complete its safety evaluation of your test program schedule, it is necessary that your application provide a sequence of performing tests during the power ascension phase. This sequential schedule should establish that the commitments of Section 14.2.11 will be met. Modify this section to provide an accurate sequence of conducting tests or present this information in a table or figure.

423.8 (14.2.12) The staff's review of your test program description disclosed that the operability of several of the systems and components listed in Regulatory Guide 1.68, Appendix A will not be demonstrated by the initial test program (although Section 14.2.7 does not list these items as exceptions). Expand your FSAR to include appropriate test descriptions for the following items from Appendix A of the guide:

A.1.c	vibration monitoring of reactor internals
A.5.d	vent and drain systems
A.5.o	shield cooling system (include reactor cavity cooling system)
A.5.p	leak detection system
A.5.r	seismic instrumentation
A.6.a	normal distribution system
A.7.b	reactor building tests (include normal containment ventilation system)
A.8	gaseous radioactivity removal systems
A.9.a	ECCS expansion and restraint tests
A.10.b	refueling equipment (hand tools and power equipment)
A.11	reactor components handling system (include containment polar crane)
A.12.b	personnel monitor and survey instruments
A.12.c	laboratory equipment
A.13	radioactive waste systems
B.1.d	post fuel loading RCS leak test
B.1.j	vibration monitoring of reactor internals
C.1.i	chemical tests to demonstrate ability to analyze and control water quality
D.1.f	effluent radiation monitoring systems
D.1.n	capability of instrumentation to detect a dropped CEA and initiate associated automatic actions
D.1.p	vibration monitoring of reactor internals

423.9
(12.2.12)

Our review of the listing of test description disclosed that several plant features which are assumed in Chapter 15 to limit or mitigate the results of postulated accidents may not be tested preoperationally. Provide test descriptions (or reference or revise existing test descriptions) which verify the operability of the following:

(1) atmospheric steam dump valves

(2) reactivity computer

(3) de-energization of pressurizer heaters on low level

(4) containment sump and associated instrumentation and alarms

(5) radwaste area sump and associated instrumentation and alarms

423.10
(14.2.12)

The staff's review of your individual test descriptions disclosed that the information provided in several of these descriptions is not sufficient for the staff to conclude that adequate testing will be performed on the systems and components covered. Expand and/or modify the test abstracts to provide the following:

(1) Test No. 1: Modify the description to provide assurance that the test will verify that emergency loads will not exceed battery sizing assumptions and demonstrate that supplied loads will operate in accordance with design requirements at minimum design voltage level (at the battery terminals) for the system.

(2) Test No. 8: Modify the test description to provide assurance that the test will verify components during the low pressure operation on the nitrogen supply system.

(3) Test No. 14: Modify the description to provide assurance that the test will verify the capability of the turbine driven feedwater pump to start and operate under the full design range of steam pressures (1210-65 psia).

(4) Test No. 20: The acceptance criteria for this test should include conformance with the plant's technical specifications limits.

(5) Test No. 22: Verify that this test or other tests, as appropriate includes the station effluent radiation monitors.

(6) Test No. 32: The acceptance criteria for this test should be expanded to include conformance with the limits that will be included in the technical specifications for the facility.

(7) Test No. 45: Modify the description to provide assurance that the test will verify that injection valves are set to prevent pump runout.

(8) Test No. 52: State whether this test is performed with steam.

(9) Test No. 54: Modify the description to provide assurance that the test will demonstrate that low pressurizer level de-energizes the pressurizer heaters.

(10) Test No. 56: Modify the description to provide assurance that the test will demonstrate, by sample analysis, the capability to purge the quench tank and establish a nitrogen blanket.

(11) Test No. 58: State how corrections will be made to account for setting pressurizer safety valve lift-points with nitrogen at ambient temperature rather than with steam at normal operating temperature. Provide supporting technical justification for the correlations.

(12) Test No. 67: Modify the description to provide assurance that the test will verify proper electrical operation of the moveable incore detector system.

(13) Test No. 75: Modify the description to provide assurance that the test will verify that the travel time of each CEA in the dashpot region satisfies mechanical design requirements and satisfies reactivity assumptions.

(14) Test Nos.76,77: Modify the description to provide assurance that the test will demonstrate that sampling procedures are adequate.

(15) Test Nos. 82, 83, 84, 85, 86, 87, & 89: For each test, state quantitative acceptance criteria and the basis for the criteria.

(16) Test No. 90: State what transients will be performed or analyzed as a part of this test, at what power level each of the transients will be conducted, the mode of operation (automatic or manual) of control systems, and provide acceptance criteria based on the predicted results of each transient.

(17) Test No. 91: State what transients and trips will be performed or analyzed to determine proper operation of control systems; state which parameters are monitored during normal operations, trips, and transients that are used to make this determination; state the mode of operation of control systems for each transient and trip; and clarify "acceptable range" of the monitored parameters in the acceptance criteria.

(18) Test Nos. 93, 94: Identify the variables or parameters to be monitored for each test; provide assurance that the test results will be compared with predicted results for the actual tests to be run (for each trip); establish quantitative acceptance criteria and the basis for the required degree of convergence of actual test results with predicted results for the monitored variables or parameters for each trip, and establish acceptance criteria for grid stability, voltage, and frequency following the generator load rejection trip.

(19) Test No. 95: Modify the description to provide assurance that the test will verify that the reactor scram will be initiated from outside the control room and that offsite power will be available.

(20) Test Nos. 99, 100: For each procedure state quantitative acceptance criteria and the bases for the criteria.

423.11 (14.2) Provide a description of testing planned to demonstrate (1) the operability and adequacy of the CEDM cooling units and the ventilation systems for the auxiliary feedwater pump rooms; (2) the capability of the reactor cavity cooling system to maintain concrete temperature below 150° F; and (3) that the insulation provided for the high temperature containment penetrations prevents excessive heating of the concrete surrounding the penetrations.

423.12 (14.2) Tests of normal and emergency ventilation systems for areas housing engineered safety features components should demonstrate that the temperature of each compartment can be maintained below the design temperature limits of the components within the compartment with the maximum expected heat load being produced in the compartment. Modify the necessary test descriptions to include this demonstration.

423.13 (14.2) Provide a description of the testing planned to demonstrate that the response time of each protection channel and ESF channel from the measured variable to the final actuating device is within the assumptions used in the accident analysis.

423.14 (14.2) Identify any of the post-fuel loading tests described in Section 14.2.12 which are not essential towards the demonstration of conformance with design requirements for structures, systems, components, and design features that:

(a) will be relied upon for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period;

(b) will be relied upon for safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions, and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions;

(c) will be relied upon for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications;

(d) are classified as engineered safety features or will be relied upon to support or assure the operations of engineered safety features within design limits;

(e) are assumed to function or for which credit is taken in the accident analysis for the facility (as described in the Final Safety Analysis Report); and

(f) will be utilized to process, store, control, or limit the release of radioactive materials

423.15 (14.2) Acceptable justification is not presented for performing the turbine trip test at 80% power instead of at 100% power as stated in Regulatory Guide 1.68. It is also not clear that the generator trip test meets the intent of the guide. The reasons for including the generator trip test in Regulatory Guide 1.68 were to assure that the turbine generator would not exceed its design speed and to establish that the plant's electrical system would perform as designed for this transient test during which the system may be subjected to frequencies in excess of 60 Hz. To accomplish the test objectives, the generator should be disconnected from the transmission system in a manner that will result in the calculated maximum overspeed condition. Normally, this is accomplished by opening of the generator output breaker in a manner that will require a turbine generator overspeed condition to initiate closure of the steam admission or stop valves.

It is our understanding that typical designs of the trip logic for the generator output breakers will, for certain sensed plant conditions, result in a direct and simultaneous trip of the turbine stop valves. There usually are additional trips that will also open the generator output breakers without directly tripping the turbine stop valves. Therefore, the latter type of trip should be simulated to initiate the transient.

Modify Section 14.2.7 and the test descriptions as necessary to clarify that the generator trip test will be performed as intended by Regulatory Guide 1.68 and to either state that the turbine trip test will be performed at 100% power or provide technical justification for conducting the test at a different power level.

423.16 Provide more detailed test descriptions for testing the core protection calculator, CEA calculator, and core operating limits supervisory system both preoperationally and following fuel loading.

423.17 (14.2.12)	Provide a test description for vibration monitoring performed at 20, 50, 80, and 100% power as listed in Table 14.2-2.
423.18 (14.2.7)	Provide additional justification for omitting or reducing in scope low power physics tests and power ascension tests on Unit No. 3. Also, for the tests that will be reduced in scope, describe what will be modified or omitted from these tests.
423.19 (15.2)	Provide a description of the testing planned to demonstrate the operability of any structures, design features, or systems to protect the facility from external and internal flooding (include leak tightness tests of compartments, doors, and waterproof hatches of safety equipment areas).

MAY 2 4 1977

Docket Nos. 50-361
and 50-362 ✓

MEMORANDUM FOR: San Onofre Units 2 and 3 Technical Reviewers

FROM: M. M. Mlynczak, Project Manager, Light Water Reactors Branch No. 2, DPM

Original signed by
K. Kniel

THRU: Karl Kniel, Chief, Light Water Reactors Branch No. 2, DPM

SUBJECT: SAN ONOFRE 2 AND 3 OPERATING LICENSE REVIEW

You are reminded that Q-1's for the San Onofre Units 2 and 3 review are due Friday, June 3, 1977. Any questions or problems you might have should be brought to my attention or to that of Harry Rood, ext. 27701.

Original Signed by

M. M. Mlynczak, Project Manager
Light Water Reactors Branch No. 2
Division of Project Management

Distribution

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SURNAME	MMMlynczak:mt	HRood	KKniel			
DATE	5/ /77	5/ /77	5/ /77			

MAY 1 7 1977

Docket Nos. 50-361
and 50-362

MEMORANDUM FOR: Karl Kniel, Chief, Light Water Reactors Branch No. 2, DPM

FROM: Harry Rood, Project Manager, Light Water Reactors Branch No. 2, DPM

SUBJECT: FORTHCOMING MEETING WITH SOUTHERN CALIFORNIA EDISON COMPANY ON SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DATE & TIME: Wednesday, May 18, 1977 - 1:00 p.m.

LOCATION: Room P-114
Bethesda, Maryland

PURPOSE: Discuss meteorological tracer tests.

PARTICIPANTS: SOUTHERN CALIFORNIA EDISON
(W. Moody, et al)

NRC - STAFF
(J. Goll, E. Markee, H. Rood)

Original Signed by

Harry Rood
Light Water Reactors
Branch No. 2
Division of Project Management

M 4
6D

OFFICE➤	DPM:LWR #2					
SURNAME➤	HRood:mt					
DATE➤	5/ /77					

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

MAY 3 1977

Docket Nos. 50-361
and 50-362

MEMORANDUM FOR: William H. Regan, Jr., Chief
Environmental Projects Branch 2, DSE

FROM: Robert B. Samworth, Section Leader
Aquatic Resources Section
Environmental Specialists Branch, DSE

SUBJECT: DRAFT QUESTIONS FOR SAN ONOFRE UNITS 2 & 3
OL STAGE REVIEW

PLANT NAME: San Onofre Nuclear Generating Station, Units 2 & 3
LICENSING STAGE: OL
DOCKET NUMBERS: 50-361 and 50-362
RESPONSIBLE BRANCH: Environmental Projects Branch 2
PROJECT MANAGER: Oliver D. T. Lynch
DATE REQUEST RECEIVED BY ESB: April 11, 1977
REQUESTED COMPLETION DATE: April 27, 1977
DESCRIPTION OF RESPONSE: Draft Q's to EPM
REVIEW STATUS: Aquatic Resources Section Review - Complete

We have reviewed the applicable aquatic resources related sections of the San Onofre ER (OL Stage) and provide the attached Draft Q's for the site visit agenda. These specific questions are in addition to those raised in our ER (OL) Acceptance Review which were transmitted by memorandum dated January 5, 1977.

This review was conducted by R. Romano.

Original Signed by Robert B. Samworth

Robert B. Samworth, Section Leader
Aquatic Resources Section
Environmental Specialists Branch, DSE

DISTRIBUTION
Docket Files 50-361
and 50-362
NRR:Rdg
ESB:Rdg
ET:Rdg

Attachment:
Draft Q's

cc: w/attachment
See attached

MY

OFFICE ➤	DSE:ET:ESB	DSE:ET:ESB				
SURNAME ➤	RRomano:jd	RBSamworth				
DATE ➤	5/ /77	5/ /77				

cc: w/attachment
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D. Muller
M. Ernst
V. Moore
R. Ballard
R. Samworth
R. Romano
O. Lynch
W. McDonald

SAN ONOFRE - OL REVIEW QUESTIONS

2.1.4.4 Surface and Ground Water Use

Provide a diagram of the sanitary waste water system, showing the routing of this waste through Unit 1 SONGS.

Table 2.1-36 SONGS Units 2 & 3 Water Use

Indicate what will be the fate of building and parking lot storm water runoff. Will waste treatment of storm water occur, if so explain the treatment, if no treatment occurs what is the fate of the untreated runoff?

2.2.2 Aquatic Ecology

Identify the location of the discharge of SONGS Units 2 & 3 in relationship to the intertidal zone.

2.2.2.3 Plankton Communities

Provide any insite into seasonal fluctuations in phytoplankton near SONGS 2 & 3 and will plant operation be expected to cause a change in dominant species.

2.2.2.3 Plankton Communities

Indicate the availability of primary productivity measurements during the pre-operational period for SONGS 2 & 3.

2.2.2.5 Pre-existing Environmental Stresses

In this section it is stated there are a variety of pre-existing environmental stresses being subjected to SONGS 2 & 3 besides high turbidity, clammers and erosion what are the other stresses.

Industrial and municipal sources were neglected. Identify such sources which may effect water quality within a five-mile radius of SONGS 2 & 3.

2.4.1 Site Vicinity Hydrosphere Description

How often do San Onofre and San Mateo Creek contain water? What is the flow of each?

2.4.3.5 Temperature

Provide the readings of water temperature in the vicinity of the intake structure over all seasons.

2.4.3.7 Dissolved Oxygen, pH and Coliform

Provide the nearshore dissolved oxygen concentration below the thermocline.

2.4.3.7 Dissolved Oxygen, pH and Coliform

Provide a list of water quality standards as required by the California Regional Water Quality Control Board.

2.3.4.8.2 SONGS Turbidity Situation

The operation of SONGS Unit 1 produces a visible turbidity plume. How will the operation of SONGS Unit 2 & 3 increase this plume?

At what depths of water are the intake and discharge of SONGS Units 2 & 3? Are these intake and discharge structures located below the thermocline during any season?

3.3.5 Make-up Demineralizer System

Provide an explanation of how the demineralizer area discharge will be neutralized to (pH 6-9) before emitted to the ocean.

3.3.6 Blowdown Processing System

Indicate what other water quality parameters will be investigated for blowdown discharge other than pH.

3.3.10 Circulating Water System

Trash from the screens will be accummulated in baskets and removed from the site to a legal point of discharge. Identify this point of discharge.

3.3.13 Main Condenser

Provide the metalic composition of the condenser tubing.

Indicate the amount of Sodium hypochlorite solution that is injected into the circulating water pump to control marine organisms; what is the expected total residual chlorine of this effluent?

3.3.14.2 Average Water Flows at Minimum Power Operation

Describe environmental impact to receiving water expected in the event of an abnormal case of air in leakage or condenser leakage when the hydrazine feed and blowdown processing system regeneration frequency may be increased.

3.3.14.2 Average Water Flows at Minimum Power Operation

Indicate the fate of the borate ion used in this process.

3.4.1 Circulating Water System

Indicate for Units 2 & 3 the distance from the first and last diffuser discharge port to the shore and to the surface of the water.

3.6.2.2 Overboard System

Indicate where the overboard water system discharge will be located. What is the volume of the condensate sumps and what is the "overboarding" rate?

3.6.2.3 Air Ejector Condenser

Explain the possible environmental effects caused by high pH in the air ejector condenser.

3.6.2.4 Blowdown Processing System (BPS) Neutralization Sump Effluent

Indicate the expected levels of iron and copper in the blowdown crud. Also indicate where this material will be deposited.

3.6.2.5 Make-up Demineralizer Neutralization Sump Effluent

Indicate the time duration frequency, and fate of this discharge. It is stated that a release of 2000 ppm sodium and 4200 ppm sulfate ions in this process is expected. What is the cfs of the discharge?

3.6.2.6 Steam Generator Blowdown

Provide the concentration of iron and copper in this discharge, give the frequency, time duration, and the cfs of the discharge.

3.6.2.7 Steam Generator Blowdown Demineralizer Effluent

This section states "the concentration of any pollutants in this effluent is expected to always be lower than the EPA Guidelines". Indicate the expected pollutants and their concentration.

3.6.2.8 Oil Removal System

Indicate the expected frequency of the oil removal system becoming inoperable or upset.

3.6.2.9 Storm and Roof Drains

Indicate the percentage of onsite storm water that will be treated by SONGS.

5.1.2.3 Effects of Local Eddy Currents

In this section it is stated that during the worst case the thermal plume would be 11 miles in diameter; what would be the distance of the plume

from the discharge structure and what would be the expected plume temperature above ambient water temperature.

Table 5.3-1 Chlorine Measurement

Give location of sampling values in the column labeled "Receiving Waters".

5.3.2 Other Chemicals

Provide a list of corrosion inhibitors with expected concentrations.

5.4 Effects of Sanitary Waste Discharge

Can the STP of Unit 1 be expected to handle successfully the waste water from Units 2 & 3. Will there be any expected water quality deterioration to the receiving water from this operation?

5.4 Effects of Sanitary Waste Discharge

What wastes will be discharged into the STP? Give average flow rates.

6.1.1 Marine Biological Monitoring Program

Explain why such an extensive biological and water quality intertidal monitoring program is being proposed. What environmental effects do you expect the operation of SONGS 2 and 3 to impose on the intertidal zone?

6.1.1 Marine Biological Monitoring Program

Justify why Zone 6 will be used as a control station. Will currents ever carry the plume to this station

6.1.2.2 Turbidity

Provide data on turbidity (suspended solids) from the intake and discharge of SONGS Unit 1 and expected levels of Units 2 and 3.

6.1.2.3 Other Chemicals

Indicate if the heavy metal levels will be reported as total metal or suspended and dissolved.

6.2.1 Proposed Operational Monitoring Program - Heavy Metals

Explain why samples for heavy metal analysis in water and sediment will not be taken from the discharge or next to the discharge.

6.2.1 Proposed Operational Monitoring Program - Heavy Metals

Justify why heavy metal levels will be taken for Cr, Cu, Ni and Fe at Unit 1 (Section 6.1.2.3) and only for Fe at Units 2 and 3.

6.2.2 Dissolved Oxygen

D. O. measurements will be taken in Zones 1B, 0B, and 6: in which zone is the discharge of Unit 1, Unit 2, and Unit 3?

6.2.3 Turbidity

Indicate the direction from the plant of the plume from SONGS 1, 2 &3 over all seasons.

MAY 2 1977

Docket Nos. 50-~~561~~ 361
and 50-~~562~~ 362

MEMORANDUM FOR: William H. Regan, Jr., Chief
Environmental Projects Branch 2, DSE

FROM: Robert B. Samworth, Section Leader
Aquatic Resources Section
Environmental Specialists Branch, DSE

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATIONS (SONGS)
UNITS 2 & 3 PREOPERATIONAL MONITORING PROGRAM

In response to a request from O. D. T. Lynch, EPM for the SONGS Units 2 and 3, dated April 7, 1977, we have reviewed the applicant's proposed Preoperational Monitoring Program. We have determined that all substantive questions on the proposed program have been raised previously in our ER (OL) Acceptance Review and are included as Draft Q's for the site visit scheduled for the week of May 16, 1977. In discussing this determination with the EPM, we reached agreement that all questions raised in our review could best be treated in conjunction with the site visit agenda. Following the site visit, a staff position will be developed by the section on the acceptability of the proposed program.

This review was conducted by C. Billups and R. Romano of this section. The latter staff reviewer will represent the Environmental Specialists Branch on the site visit.

/s/ C. Billups

for Robert B. Samworth, Section Leader
Aquatic Resources Section
Environmental Specialists Branch, DSE

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cc: H. Denton
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M. Ernst
V. Moore
R. Ballard
O. Lynch
R. Samworth
C. Billups
R. Romano

my

OFFICE	DSE:ET:ESB	DSE:ET:ESB	DSE:ET:ESB
SURNAME	CWBillups:jd	RRRomano	RBSamworth
DATE	5/ /77	5/ /77	5/ /77

MAY 2 1977

Docket Nos. 50-361
and 50-362

MEMORANDUM FOR: Karl Kniel, Chief, Light Water Reactors Branch No. 2, DPM

FROM: Harry Rood, Project Manager, Light Water Reactors Branch No. 2, DPM

SUBJECT: FORTHCOMING MEETING WITH SOUTHERN CALIFORNIA EDISON COMPANY AND SAN DIEGO GAS AND ELECTRIC COMPANY ON SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NOS. 2 AND 3

Date and Time: Wednesday, May 4, 1977 - 9:00 a.m.
Thursday, May 5, 1977 - 9:00 a.m.

Location: Wednesday: Construction Administration Bldg, San Onofre Nuclear Generating Station, San Clemente, California
Thursday: Same as Wednesday

Purpose: Site Visit: To discuss seismic and structural audit of Category I structures, and inspect such structures at Units 2 and 3.

Participants: Southern California Edison

D. Hayden, et al

NRC

R. Lipinski, S. Chan, H. Rood

Original Signed by

Harry Rood
Light Water Reactors
Branch No. 2
Division of Project Management

M 4
OD

OFFICE➤	DPM:LWR #2					
SURNAME➤	HRood:ab					
DATE➤	5/ /77					

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A. Meltz

APR 6 1977

Docket Nos. 50-361
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MEMORANDUM FOR: San Onofre Nuclear Generating Station Units 2 & 3 Technical Reviewers (See Distribution List)

FROM: Harry Rood, Project Manager, Light Water Reactors Branch No. 2, DPM

THRU: Karl Kniel, Chief, Light Water Reactors Branch No. 2, DPM

SUBJECT: PRELIMINARY SAN ONOFRE UNITS 2 & 3 OPERATING LICENSE REVIEW SCHEDULE

Enclosed for your information is the preliminary schedule for the San Onofre Units 2 & 3 OL review. I am forwarding it to you prior to final approval to advise you as early as possible of the anticipated review timetable.

Original signed by
Harry Rood, Project Manager
Light Water Reactors Branch No. 2
Division of Project Management

Enclosure:
Preliminary San Onofre
2 & 3 Review Schedule

OFFICE	DPM:LWR #2	DPM:LWR #2	DPM:LWR #2			
SURNAME	MMlynczak:ab	HRood HR	KKniel			
DATE	4/6/77	4/5/77	4/6/77			

APR 6 1977

ENCLOSURE

PRELIMINARY SAN ONOFRE 2 & 3 OL REVIEW SCHEDULE

MILESTONE	Δ WKS	ELAPSED WKS	DATE
Docketing	0	0	3/23/77
Applicant Acceptance Review Responses	5	5	4/29/77
Q-1's to LPM	5	10	6/3/77
Q-1's to Applicant	3	13	6/24/77
Appl. Replies to Q-1's	8	21	8/19/77
Staff Positions to LPM	8	29	10/14/77
Staff Positions to Appl.	3	32	11/4/77
Appl. Resp. to Staff Pos.	8	40	12/30/77
SER Input to LPM	8	48	2/24/78
Draft SER By LPM	4	52	3/24/78
Mgt. Review Draft SER	2	54	4/7/78
SER Issued	2	56	4/21/78
ACRS Meeting	7	63	6/9/78
ACRS Letter	1	64	6/16/78
SER Supplement	8	72	8/11/78

Hearing Schedule (Preliminary)

Earliest Start Rad/Safety Hearing	9/12/78
End Rad/Safety Hearing	11/10/78
ASLB Decision	1/19/79

Licensing Effort Complete (1 month prior to fuel load)	1/80 (Unit 2) 4/81 (Unit 3)

PDD (Fuel Load Date)	2/80 (Unit 2) 5/81 (Unit 3)

OFFICE ➢						
SURNAME ➢						
DATE ➢						

NRC FORM 318

Docket

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

FEB 24 1977

DOCKET NOS. 50-361
AND 50-362

APPLICANTS: SOUTHERN CALIFORNIA EDISON COMPANY AND SAN DIEGO GAS AND ELECTRIC COMPANY

FACILITY: SAN ONOFRE NUCLEAR GENERATING STATIONS UNITS 2 AND 3

SUBJECT: SUMMARY OF REACTOR VESSEL SUPPORT MEETING ON FEBRUARY 10, 1977

Enclosed is a summary of a meeting held with Southern California Edison Company and Combustion Engineering. The purpose of the meeting was to obtain the information necessary to perform an analysis of the San Onofre 2 reactor vessel supports.

Harry Rood

Harry Rood, Project Manager
Light Water Reactors Branch No. 2
Division of Project Management

cc: See next page

cc: Southern California Edison Company
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Mr. W. D. Griffith
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San Diego, California 92112

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

FEB 11 1977

MEMORANDUM FOR: R. J. Bosnak, Chief, Mechanical Engineering Branch, DSS

FROM: R. K. Mattu

THRU: H. L. Brammer, Section Leader HLB

SUBJECT: MEETING WITH COMBUSTION ENGINEERING AND SOUTHERN CALIFORNIA EDISON COMPANY TO OBTAIN INFORMATION FOR ANALYSIS OF REACTOR VESSEL SUPPORT DESIGN OF SAN ONOFRE 2 NUCLEAR POWER PLANT

The Mechanical Engineering Branch of NRC intends to develop a model for the 3410 MW(t) CE nuclear steam supply system and perform an independent North Anna analysis. To achieve this goal a meeting was conducted at Combustion Engineering plant at Windsor, Connecticut. Participants were NRC, INEL, CE and Southern California Edison Company whose San Onofre 2 plant has been selected for this analysis. A list of participants from these organizations is enclosed as Attachment 1. Modeling information that was required is indicated in Attachment 2.

NRC at the beginning of the meeting stated its goals and the need for the information. Mr. D. Hayden of SCE and T. Natan of CE assured NRC that they will provide us all the information that we might need to develop the model and perform the analysis. Most of the drawings were obtained on the spot (Attachment 3) and some interface drawings (CE & Bechtel) will be sent to NRC by March 1, 1977. SCE has asked CE to provide NRC with time history forcing functions (based on cold leg break and hot leg break) that are essential for the analysis. Attachment 4 has a short description of the dynamic analysis that MEB/NRC proposes to perform.

RKMattu
R. K. Mattu
Mechanical Engineering Branch
Division of Systems Safety

Attachment

Meeting on "North Anna" Analysis Info 2/10/77

Art SWAN	CE	5123
R.K. MATTU	NRC	(301) 492-7538
BF Saffell (INEL)	EG&G Idaho	
DR Hayden	SCE	
EJ GERLOFF	SCE	
Dan Peck	CE	CE ext 3772
T.E. Natan	"	" 5709
D.W. Tolli	"	" 2967
Fred Martson	"	" 3102
PAUL WYSOCKI	"	" 4113
W. Borchers	"	" 3417
R.P. Kassawara	CE	

Attachment 2.

MODELING INFORMATION FOR COMBUSTION ENGINEERING NUCLEAR STEAM SUPPLY SYSTEM

INFORMATION	TECHNICAL RESPONSIBILITY
I. STRUCTURAL DETAILS (BLUEPRINTS) OF:	
A. Reactor Vessel	
1. Dimensions (heights, radii, thicknesses, head connectors, nozzle details)	CE
2. Materials	CE
B. Upper Core Support Structure (Support Columns, Upper Support Plate (including hole pattern), Internal Support Ledge, Hold-down Spring)	
1. Dimensions	CE
2. Materials	CE
3. Details of interfaces between U.S. plate, ledge, core barrel, reactor vessel. Include equivalent springs used in analysis	CE
C. Core Barrel no thermal shield	
1. Dimensions	All Internals are 304SS CE
2. Materials	CE
3. Nozzle details	CE
4. Details of interfaces with:	
a. ~~Thermal shield~~	NA
b. Upper and Lower Core Plates	CE
c. Radial Supports	CE
d. Bottom Support Casting	CE
e. Baffles (Flow Shroud)	CE
f. RPV Nozzles	CE
D. Upper and Lower Core Plates	
1. Dimensions (including hole patterns)	CE
2. Materials	CE
3. Details of interfaces with	CE
a. Support Columns	
b. Fuel Assemblies	
c. Baffles	
E. Thermal Shield — NO	NA
1. Dimensions	
2. Materials	

INFORMATION	TECHNICAL RESPONSIBILITY
I. STRUCTURAL DETAILS (BLUEPRINTS) OF: (Cont'd.)	
F. Bottom Support (Columns and Bottom Support Casting)	
1. Dimensions	CE
2. Materials	CE
3. Details of Interfaces with	CE
a. Each other	
b. Lower core plate	
G. Piping	
1. Materials	CE
2. Complete plant layout including dimensions, elevations, angles, supports prim coolant piping	Bechtel
H. Reactor Vessel Supports	
1. Dimensions	Bechtel/CE
2. Materials	Bechtel/CE
3. Details of sliding support restraint and equivalent stiffness of base structure	Bechtel/CE
Bolts on Pads are Preloaded	
.67g **I. Steam Generator**	
1. Dimensions	CE
2. Materials	CE
3. Details of interfaces with snubbers stiffeners and S.G. support	Bechtel/CE
4. Steam line and feedwater line details with hangers and snubbers detailed (location and stiffness)	Bechtel
J. Pump	
1. Dimensions	CE
2. Materials	CE
3. Details of interface with pump support	Bechtel/CE
K. Pressurizer	
1. Dimensions	CE
2. Materials	CE
3. Details of interface with support	Bechtel/CE

INFORMATION	TECHNICAL RESPONSIBILITY
I. STRUCTURAL DETAILS (BLUEPRINTS) OF: (Cont'd.)	
L. Steam Line	
1. Dimensions } see I.4.	Bechtel
2. Materials	Bechtel
M. Component Supports	
1. Steam Generator (including snubbers)	
a. Dimensions	CE
b. Materials	Bechtel/CE
c. Details of connectivity	Bechtel/CE
d. Spring rates where applicable (snubbers or columns)	CE
2. Pump (including snubbers) (Same as I.M.1.a.-d.)	Bechtel/CE
3. Pressurizer (Same as I.M.1.a.-d.)	Bechtel/CE
4. Valves if applicable no valves in Prim. loop	NA
5. Any other restraints needed (pipe whip, hangers, etc.) gaps, location, stiffness based on D.B. breaks	Bechtel
N. Biological Shield	
1. Dimensions and materials	Bechtel
2. Connectivity if any	Bechtel
II. MASS PROPERTIES	
A. Masses, C.G. Locations, Moments of Inertia for:	
1. Reactor vessel	CE
2. Bottom support casting	CE
3. Fuel assembly	CE
4. Control Rods and Drivers	CE
5. Reactor and Internals Total (wet or dry)	CE
6. Valves — None	NA
7. Steam Generator	CE
8. Pump	CE
9. Pressurizer	CE

Drawings Presented to NRC for North Anna Analysis (one copy of each provided)

Drawing No.	Rev	Component
E-234-590	4	Steam Generator
* E-234-592	9	Steam Generator
E-234-586	3	R.V. COLUMN
E-234-550	3	Reactor Vessel
* E-234-551	2	" "
E-234-587	4	R.V. COLUMN
* E-235-176	2	PIPE
* 2F-1565 sheet 1	0	Reactor Coolant Pump
* 2F-1565 sheet 2	0	" " "
E-1370-320-007	04	S.G. Supports
E-1370-320-022	04	" "
* E-1370-320-035	01	R.V. supports
PD 16154	0	Reactor Coolant Pump Snubber
D-1370-320-052	03	RC PUMP SKIRT
D-1370-320-053	03	RC PUMP supports
D-1370-320-055 sht 1	05	" " "
D-1370-320-056	05	" " "
D-1370-320-054	04	" " "
D-1370-320-055 sht 2	05	" " "
D-1370-320-057	01	" " "
E-1370-320-058 sht 1	01	R.C. PUMP Installation
E-1370-320-058 sht 2	01	" " "

Drawing No		Rev	Component
E-1370-320-059		00	R.C. Pump Installation
E-234-981		7	Pressurizer
* E-234-982		6	"
CE CALCULATION 3400-16-40-30		6/24/74	INTERNALS WEIGHTS
E1370-164-311		02	CORE SUPPORT BARREL
E-STD-164-313		05	" " "
E-STD-164-314		04	" " "
E-STD-164-315		04	" " "
E-STD-164-316		06	" " "
E-1370-164-303		02	Reactor Internals
E-STD-164-326	sht 1	04	CORE SHROUD
"	sht 2	04	" "
"	sht 3	04	" "
E-STD-164-312	sht 1	07	CORE PLATE
"	sht 2	07	" "
E-STD-164-332	sht 1	08	UPPER Guide Structure
"	sht 2	08	" " "
E-STD-164-335		03	CEA SHROUD
E-STD-164-333	sht 1	06	Alignment Plate
"	sht 2	06	" "
E-3072-164-331	sht 1	01	UPPER Guide STRUCTURE
"	sht 2	01	" " "
E-STD-161-017		03	FUEL
E-STD-164-001	sht 1	01	INTERNALS

Drawing No		Rev	Component
E-STD-164-001	sht 2	01	Internals
E-1370-320-036		01	RV Supports
E-235-177		2	PIPE
E-235-178		3	PIPE
E-235-179		3	PIPE
E-235-180		3	PIPE
E-1370-320-001	sht 1	08	RCS LOAD TABLE
E-1370-320-021		03	STEAM Generator Support
E-STD-220-031		05	" " "
E-STD-220-032		06	" " "

* Mr. R.K. MATTU REQUESTS ONE ADDITIONAL COPY OF

Attachment 4

Dynamic Analysis of Reactor Coolant Systems under LOCA Conditions

This is a brief description of analysis of a 3410 MW(t) CE pressurized water reactor coolant system to determine support loads due to a postulated break in the primary coolant piping. The analysis will include structural modeling, nonlinear time history dynamic structural analysis of a 3-dimensional model of the reactor coolant system including coupled details of the reactor internals, pressure vessel, supports and piping. The dynamic analysis is performed to determine the reactions of the reactor vessel supports to the simultaneous effects of pipe thrust of external and internal horizontal and vertical asymmetric pressure loads applied to the reactor vessel and internals as a consequence of the postulated pipe rupture.

System Description (3410 MW(t))

The reactor internals, including the fuel and supporting structures are suspended from the closure head flange region of the reactor vessel and are surrounded by the cylindrical "core support barrel" (CSB) as shown in Figure 1. The vessel in turn is surrounded by the biological shield wall, which forms the RV cavity.

The reactor internals are designed to support the reactor core fuel assemblies, guide the control element assemblies (CEA), absorb and transmit the dynamic loads and other loads to the reactor vessel flange and provide flow paths for the reactor coolant and guide in-core instrumentation. The major structures however are the core support barrel (CSB), fuel assemblies, upper guide structure, lower support structure and core shroud. Interfaces with RV are at vessel flange where CSB and upper structure flanges are supported and held down, at the core barrel snubbers and at the core barrel-outlet nozzle jucture. The CSB & RV are essentially concentric cylinders throughout the length of CSB. See Figure 2 for RV support arrangement. The vessel is supported on four vertical columns

located under the vessel inlet nozzles. A pad welded to each inlet nozzle provides the surface to which the column is bolted. This pad is designed to act as a radial key and to mate with the support structure. As per CE explanation, low friction bearings and tangential gaps allow face growth during thermal expansion while resisting vessel horizontal motion during earthquakes (SSE for this plant is 0.67g) and following LOCA. At the bottom of each column is a base plate which is bolted to the supporting structure. The base plate is assumed to act as a key way for a horizontal key welded to the vessel lower head. Low friction bearings and tangential gaps are provided to allow radial motion of the key while resisting motions resulting from earthquake/LOCA. A nonlinear energy absorbing material is used at the key way interface to limit the shear load applied to the vessel lower head during a LOCA.

Problem Description

Upon postulation of a break in a primary coolant pipe several rapidly occurring events cause internal and external transient loads to act upon the RV. For the RV inlet pipe break, asymmetric pressure changes take place in the annulus between the CSB and the vessel. Decompression occurs on the side of the vessel annulus nearest the pipe break before pressure on the opposite side changes. The momentary difference in pressure across the CSB induces lateral loads in opposite directions on the CSB and the RV. Vertical loads are also applied to the internals and to the vessel due to the vertical flow resistance through the core and asymmetric axial decompression of the vessel. Simultaneously, as fluid escapes thru the break, the annulus between the RV and biological shield wall becomes asymmetrically pressurized resulting in a difference in pressure across the RV causing additional horizontal and vertical external loads on the vessel. In addition, the vessel is loaded by the effects of initial tension release and blowdown thrust at the pipe break.

The loads occur simultaneously and are shown in Figure 3. For a reactor vessel outlet break the same type of loadings in Figure 3 occur but the

internal loads are predominantly vertical due to more rapid decompression of the upper plenum. For each postulated break, the time history of the RV support reactions due to the complete set of simultaneous horizontal and vertical loads will be supplied by CE.

Model

A condensed structural model of the RCS and reactor internals will be developed from the detailed drawings (Attachment 2) of components by maintaining response characteristics and interface response compatibility.

Forcing Functions

The size of the breaks that will be considered will be based on the dynamic analysis of the reactor coolant piping including motion limiting devices. Following information was supplied by CE (1) for the reactor inlet guillotine break, the location and geometry of the pump supports and pipe stop determines the break size of 350 in^2; (2) the reactor outlet break is limited to 100 in^2 by the steam generator supports and the inherent stiffness of the hot leg without pipe stops.

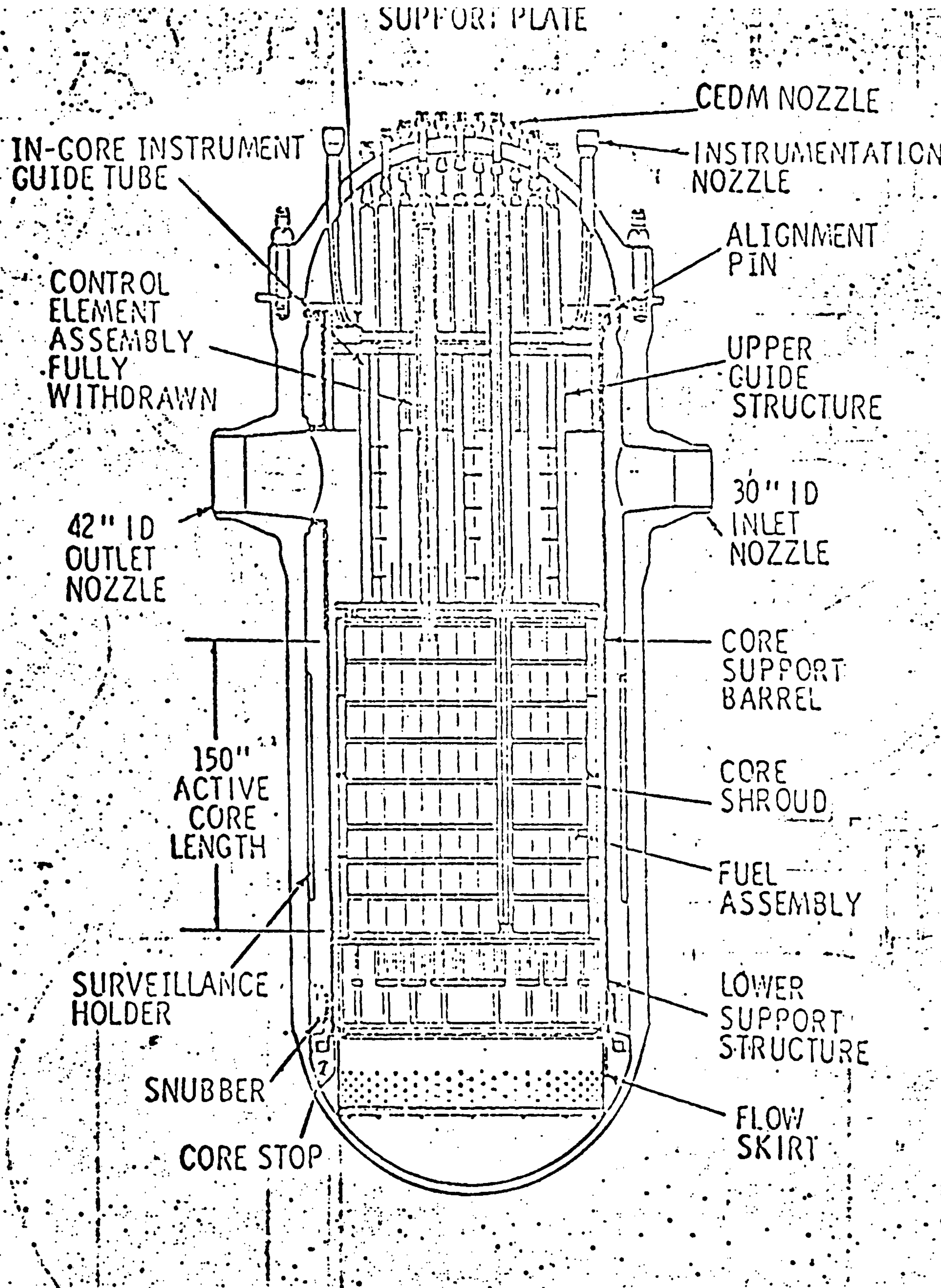
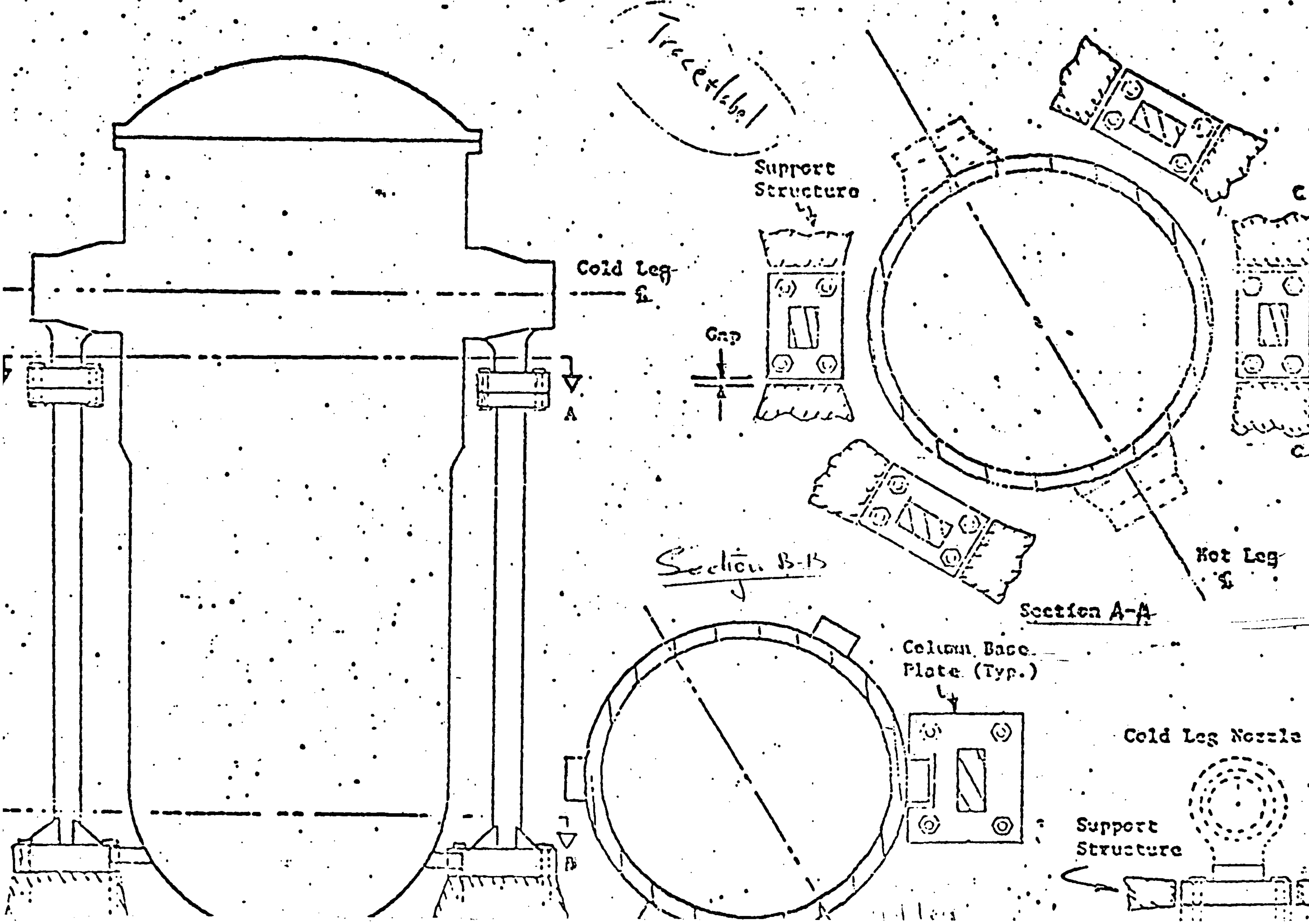


Figure 1 - Reactor Vessel Arrangement

Figure ~~12~~ - Reactor Vessel LOCA Supports



P_T = Pipe Break Thrust
P_C^H = Horizontal Cont. Press.
P_C^V = Vertical Cont. Pressure
P_I^H = Horizontal Hydraulic Force
P_I^V = Vertical Hydraulic Force
P_R^H = Horizontal Hydraulic Force
P_R^V = Vertical Hydraulic Force

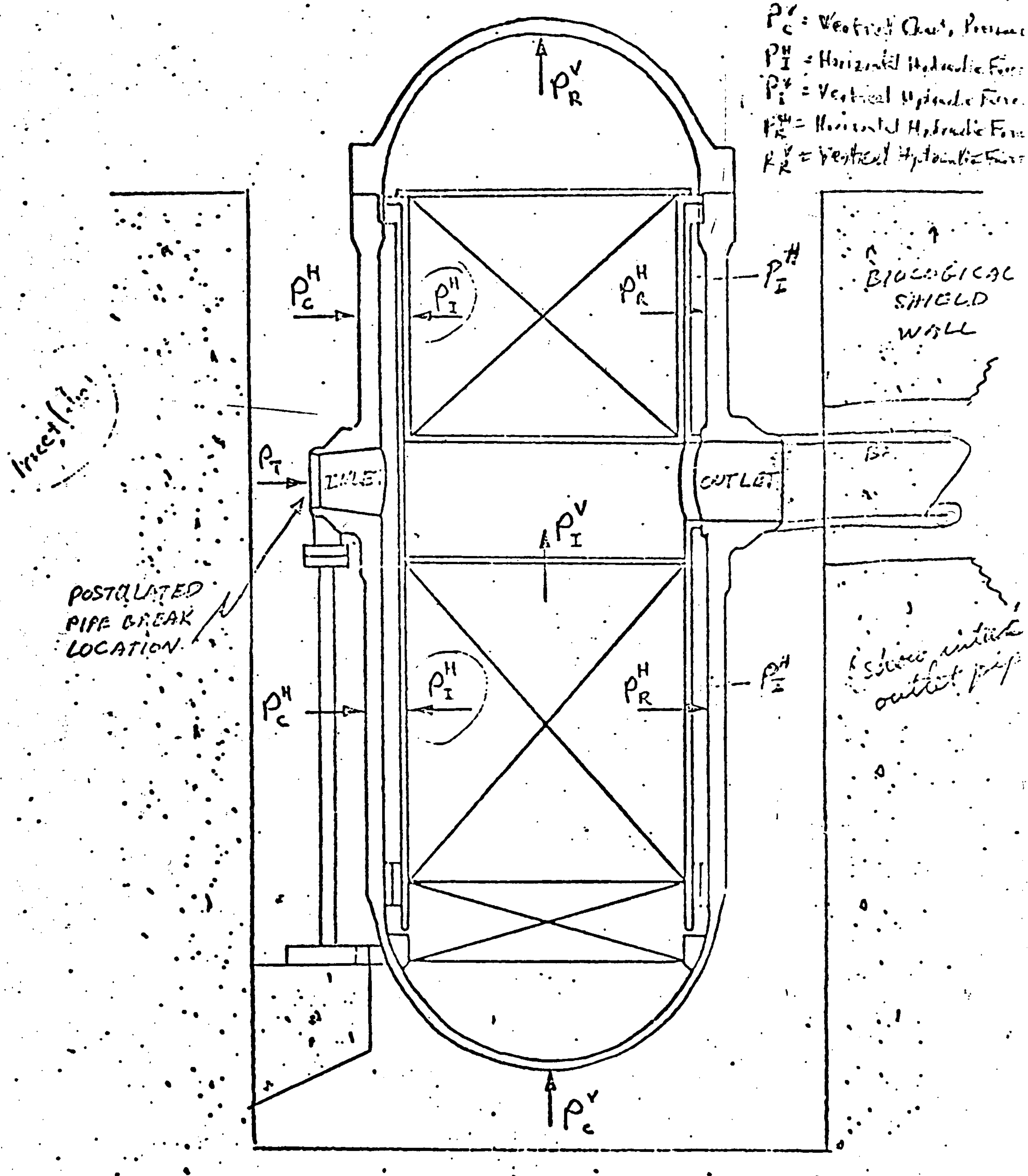


FIGURE 3 [illegible] Following [illegible] LOCA

JAN 2 4 1977

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LWR #2 File
DBVassallo
KKniel
LChandler
STreby
HRood
MMlynczak
VAMoore

Docket Nos. 50-361
and 50-362

MEMORANDUM FOR: Roger S. Boyd, Director, Division of Project Management

FROM: Karl Kniel, Chief, Light Water Reactors Branch No. 2, DPM

SUBJECT: STATUS OF ACCEPTANCE REVIEW, SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 & 3, FSAR

Background

On December 1, 1976, the Southern California Edison Company tendered an application for operating licenses for the San Onofre Nuclear Generating Station, Units 2 and 3. The tendered application includes the Final Safety Analysis Report in the Revision 2 Format, General Information, the Environmental Report and Antitrust Information. The acceptance review of the tendered application was initiated upon receipt.

Current Status

The staff has completed its preliminary review of the FSAR. Acceptance of the application for docketing has been recommended by all branches with the exception of Hydrology/Meteorology Branch, Hydrology Section. However, several branches recommended conditional acceptance, subject to obtaining from the applicant a commitment to supply in timely fashion all technical information not detailed in the FSAR but required for initiation of an independent review. This includes information on the Core Protection Computer (CPC), Electrical, Instrumentation and Control Drawings, Seismic Qualification data, Environmental Qualification data, Flood Protection Design from runoff and High Energy Pipe Break Analysis.

The applicant has been notified by telephone that in order to initiate the review now it would be highly desirable to provide the missing information within the next six months. Enclosure 1 provides a more detailed listing of this information and its present status as reported by the applicant or by other sources within NRC.

MEMO 4

OFFICE➤						
SURNAME➤						
DATE➤						

The applicant has submitted the FSAR for review a full three years in advance of scheduled fuel loading, arguing that resolution of the value of ground acceleration for the Design Basis Earthquake (including obtaining input from the U. S. Geological Survey) as well as detailed review and evaluation of the methods used in seismic design and the probability of an extended hearing provide sufficient bases for the immediate initiation of the review process. We believe that the applicant's arguments have merit and intend to decide the docketing issue as soon as the applicant commits to provide the necessary information on an acceptable timetable.

Staff Action

We recommend that the applicant be notified in writing of the due date for missing information via Enclosure 2. The applicant is expected to respond with an update of anticipated availability dates for all outstanding information listed in Enclosure 1. If the applicant states that certain information cannot be submitted in entirety within the six month envelope provided, the docketing decision will be made based on a judgment of the degree of completeness of the FSAR in light of revised availability dates and percentage of information still unavailable.

Original signed by
K. Kniel

Karl Kniel, Chief
Light Water Reactors
Branch No. 2
Division of Project Management

Enclosures:
1. Data Required for Docketing Decision - Current Timetable
2. Letter to Applicant Requesting Revised Timetable

OFFICE ➤	DPM:LWR #2	DPM:LWR #2				
SURNAME ➤	MMlynczak:mt	KKniel				
DATE ➤	1/ /77	1/ /77				

JAN 2 4 1977

ENCLOSURE 1

DATA REQUIRED FOR DOCKETING DECISION
CURRENT TIMETABLE

	Item	FSAR Section	Status
(1)	CPC (Core Protection Computer) detailed drawings	1.7	90% Available - to be submitted with other Section 1.7 drawings
	CPC Software	---	Software specification due to NRC 3/4/77 on ANO-2 - Core dump due to NRC 4/1/77
(2)	Electrical, Instrumentation and Control drawings	1.7	90% completed and available now. 100% available in six months.
(3)	Instrument Location Layout Drawings	1.7	75% completed and available now. Remainder become available between 5/77 and 7/78.
(4)	Bechtel Proprietary Drawings (electrical)	1.7	25% available now. 75% to become available as equipment is delivered, through 8/78.
(5)	Seismic Qualification Data - NSSS	3.10	50% SONGS data due 3/77.
	Seismic Qualification Data - BOP	3.10B	100% available by 9/77.
(6)	Environmental Qualification Data - NSSS	3.11	Will be submitted in 6-9 months.
	Environmental Qualification Data - BOP	3.11A	100% available 11/79.
(7)	Flood Protection Design	2.4	Will be available 12/77.

JAN 2 4 1977

Item	FSAR Section	Status
(8) High Energy Pipe Break Analysis	3.6	50% inside containment available in six months. 100% available late 1978.
Outside Containment	3.6	50% outside containment due 6/77 100% available 9/78

JAN 2 4 1977

Enclosure 2

Letter to Applicant Requesting
Revised Timetable

JAN 2 4 1977

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DBVassallo
KKniel
LChandler
STreby
HRood
MMlynczak
VAMoore

Docket Nos. 50-361
and 50-362

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas and Electric Company
ATTN: Mr. Jack E. Thomas
Vice President - Electric
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

ACCEPTANCE REVIEW OF SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3, FSAR

On December 1, 1976, you tendered an application for operating licenses for the San Onofre Nuclear Generating Station, Units 2 and 3. Your application included the Final Safety Analysis Report, General Information, the Environmental Report and Antitrust Information. The acceptance review was initiated upon receipt of the application.

In order to complete our acceptance review, we must determine the completeness of the FSAR as tendered. Because all technical information necessary for the completion of an effective independent evaluation of the SONGS 2 and 3 design has not been included in the FSAR, we require definition of anticipated submittal dates prior to docketing the application and subsequent initiation of the detailed review.

We require submittal dates compatible with the anticipated review schedule for each of the following items:

(1) CPC (Core Protection Computer) detailed drawings (Section 1.7); CPC software submittal;

(2) Electrical Instrumentation and Control Drawings (Section 1.7);

(3) Instrument Location Layout Drawings (Section 1.7);

(4) Bechtel proprietary drawings (Section 1.7);

OFFICE ➤						
SURNAME ➤						
DATE ➤						

(5) Seismic Qualification data, BOP (Section 3.10B);
Seismic Qualification data, NSSS (Section 3.10);

(6) Environmental Qualification data, BOP (Section 3.11A);
Environmental Qualification data, NSSS (Section 3.11);

(7) Flood Protection Design (Section 2.4);

(8) High Energy Pipe Break Analysis (Section 3.6).

Your commitment to provide the above information within six months will allow docketing of the application and initiation of the detailed review of the FSAR. If you cannot make this commitment, please provide us with the earliest date that you can practically meet for each item, and we will use those dates as a basis for making a decision regarding docketing your application at this time.

Sincerely,

Original signed by
D. B. Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

cc: See page 3

OFFICE➤	DPM:LWR #2	DPM:LWR #2	DPM:AD/LWR			
SURNAME➤	MMlynczak:mt	KKniel	DBVassallo			
DATE➤	1/21/77	1/2/77	1/2/77			

cc: Rollin E. Woodbury, General Counsel
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas and Electric Company
111 Sutter Street
San Francisco, California 94104

Mr. David Sakai
845 North Perry Avenue
Montebello, California 90640

Brent N. Rushforth, Esq.
Center for Law in the Public Interest
10203 Santa Monica Boulevard
Los Angeles, California 90067

Mr. Kenneth E. Carr
City Manager
City of San Clemente
100 Avenido Presidio
San Clemente, California 92672

Alan R. Watts, Esq.
Assistant City Attorney
City Hall
Anaheim, California 92805

Lawrence Q. Garcia, Esq.
California Public Utilities Commission
5066 State Building
San Francisco, California 94102

George Spiegel, Esq.
2600 Virginia Avenue, N. W.
Washington, D. C. 20036

Mr. W. D. Griffith
San Diego Gas and Electric Company
P. O. Box 1831
San Diego, California 92112

Mr. B. W. Colston
San Diego Gas and Electric Co,
P. O. Box 1831
San Diego, California 92112

Dockets

JAN 5 1977

Docket Nos. 50-361
and 50-362

MEMORANDUM FOR: William H. Regan, Jr., Chief, Environmental Projects Branch No. 2, DSE

FROM: Robert B. Samworth, Section Leader, Aquatic Resources Section, Environmental Specialists Branch, DSE

SUBJECT: ACCEPTANCE REVIEW FOR SAN ONOFRE UNITS 2 & 3 ENVIRONMENTAL REPORT (OL STAGE)

PLANT NAME: San Onofre Nuclear Generating Station, Units 2 & 3
LICENSING STAGE: OL
DOCKET NUMBERS: 50-361 and 50-362
RESPONSIBLE BRANCH: Environmental Projects Branch No. 2
PROJECT MANAGER: Oliver D. T. Lynch
DATE REQUEST RECEIVED BY ESB: December 7, 1976
REQUESTED COMPLETION DATE: December 30, 1976
DESCRIPTION OF RESPONSE: Acceptance of ER Recommended
REVIEW STATUS: Aquatic Resources Section Review - Complete

We have reviewed the applicable aquatic resources related sections of the San Onofre ER (OL Stage) and find them sufficiently complete to initiate the environmental review. The specific comments which are attached should be transmitted as draft questions to the applicant.

This review was conducted by J. Lehr and C. Billups.

Original Signed by

Robert B. Samworth, Section Leader
Aquatic Resources Section
Environmental Specialists Branch
Division of Site Safety and
Environmental Analysis
Office of Nuclear Reactor Regulation

Attachment:
Comments

Drep 2

cc: w/attachment
See Attached

OFFICE ➤						
SURNAME ➤						
DATE ➤						

cc: w/attachment
H. Denton
D. Muller
M. Ernst
V. Moore
O. Lynch
R. Ballard
C. Billups
J. Lehr
R. Samworth
W. McDonald

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SURNAME ➤	CWBillups:jd	JCLehr	RBSamworth			
DATE ➤	1/ /77	1/ /77	1/ /77			

SPECIFIC COMMENTS

ENVIRONMENTAL SPECIALISTS BRANCH - AQUATIC RESOURCES SECTION

ER (OL) ACCEPTANCE REVIEW - SAN ONOFRE 2 AND 3

Section 2.2.2

Identify aquatic species, appearing on either state or Federal lists of threatened or endangered species, which might be affected by operation of the San Onofre Station.

Section 2.2.2.5

Provide a copy of Figure 2.2-8 which was omitted.

Section 2.4.3.5

Provide the basis for the receiving water temperature fluctuations cited in this section.

Provide available data which describes the characteristics such as depth, thickness and gradient of the thermocline at various times of the year.

Provide the raw data or the summary reports which comprise the basis for the numerous statements on the dissolved oxygen, pH, turbidity and coliform characteristics of the waters near the SONGS site. Presently, these statements are not referenced adequately.

Provide a discussion of the existing environmental stresses on the aquatic environment near the San Onofre site.

Section 3.3

Tables 3.3-1 and 3.3-2 are missing. Provide these tables.

Indicate the plant power level referred to in Section 3.3.14.1 as a basis for Table 3.3-2 and Figure 3.3-1. Provide similar data as in these tables for the normally anticipated plant load and minimum power level anticipated for the plant under normal conditions.

Section 3.4.4

Indicate the frequency of the heat treatment of the plant cooling water system.

Section 3.4.5

Describe procedures, species and size classes of fish used in the model testing which resulted in the proposed design of the Fish Conservation system. If results have been presented in a report, provide reference or copies (if none have been submitted to the NRC).

Section 3.6.1

Clarify the meaning of "...the maximum concentration of free residual chlorine during any chlorination is less than 0.5 mg/liter in the immediate vicinity of the discharge".

The stated expected residual chlorine concentration in this section is not consistent with that discussed in Section 5.3. Clarify this apparent discrepancy.

Section 3.6.2

Indicate the water quality characteristics and quantity of the expected releases from the Overboard System.

Table 3.6-1

The waste discharge should also indicate the concentrations of expected waste products to be discharged under normal and worst case conditions.

Section 5.1.3.4.2

Provide information on the procedures used in the monitoring of ichthyoplankton entrainment for Unit 1. Provide date when results of this study are anticipated.

Section 5.3

Provide a copy of the recently issued NPDES Permit to San Onofre Units 2 and 3.

Section 6.3.1

Provide a copy of the proposed Thermal Exception Studies.

Appendix 6B

The proposed ETS do not include Fish Entrainment and Impingement Monitoring Programs. Provide justification for omission of these programs or supplement the proposed ETS with proposed studies.

Dockets

DEC 3 0 1976

Docket Nos. 50-361
and 50-362

MEMORANDUM FOR: William H. Regan, Jr., Chief
Environmental Projects Branch 2, DSE

FROM: Jerry R. Kline, Section Leader, Terrestrial Resources Section, Environmental Specialists Branch, DSE

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NOS. 2 AND 3, OL STAGE

PLANT NAME: San Onofre Nuclear Generating Station, Unit Nos. 2 and 3
LICENSING STAGE: OL
DOCKET NO.: 50-361, 50-362
RESPONSIBLE BRANCH: Environmental Projects Branch 2
PROJECT MANAGER: Oliver D. T. Lynch
DATE REQUEST RECEIVED BY ESB: December 7, 1976
REQUESTED COMPLETION DATE: December 30, 1976
DESCRIPTION OF RESPONSE: Acceptance of ER Recommended
REVIEW STATUS: Terrestrial Resources Section Review - Complete

We have reviewed the applicable terrestrial resource related sections of Chapters 2, 3, 4, 5, 6, 9, 10 and 12 of the San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, Environmental Report. We find it sufficiently complete to begin the environmental review.

Specific comments covering various sections of the ER are enclosed. This review was conducted by G. Gears.

Original Signed by Ronald Ballard

for Jerry R. Kline, Section Leader
Terrestrial Resources Section
Environmental Specialists Branch
Division of Site Safety and
Environmental Analysis

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COMMENTS

Section 4.2.1 Southern California Edison Company - Transmission Lines

Provide the staff with a copy of all Southern California Edison's (SCE) standard specifications relating to siting, construction and operating procedures employed to avoid and/or mitigate transmission system environmental impacts.

Section 4.2.2.1 Structures (SDG&E)

Provide the staff with a copy of "SDG&E Foreman's Guide for Improving the Appearance of Transmission Lines" and all other SDG&E Guidelines or Specifications relating to siting, construction and operation of transmission lines employed to avoid and/or mitigate environmental impacts.

Section 5.5 Effects of Operation and Maintenance of the Transmission System

Specify design characteristics and intended mitigative actions to be utilized to minimize any effects on radio and television reception due to transmission system operation.

Indicate the maximum design ground level field gradients for all lines.

State the exact types of right-of-way maintenance procedures to be used. If any herbicides are to be used, indicate this fact as well as formulation and use compliance with State and Federal registration requirements.

Indicate by topographic maps all habitats along the proposed transmission line ROWs classified by State and Federal Authorities as being critical in endangered, rare, threatened or protected wildlife and plant species.

Indicate how much prime or unique farmlands (land in capability Class I, most of Class II, and Class IIIw that has an adequate water management system such as pivot irrigation - Refer to the Soil Conservation Service's Land Inventory and Monitoring Memorandum-3, October 15, 1975) will be affected by the proposed transmission siting.

Section 10.9 Alternative Transmission Facilities

Construction of the proposed transmission routes is scheduled to begin September 1977. Indicate all changes to siting and design of the proposed system since submittal of the Construction Permit Environmental Report. Provide the staff with a description of the selection method used to determine the proposed routes and provide a map showing all alternative routes considered in the selection process.

Docket Files

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

DEC 2 2 1976

Docket Nos. 50-361
50-362

MEMORANDUM FOR: Karl Kniel, Chief, Light Water Reactors Branch No. 2, DPM

FROM: Victor Benaroya, Chief, Auxiliary and Power Conversion Systems Branch, DSS

SUBJECT: ACCEPTANCE REVIEW - SAN ONOFRE NUCLEAR GENERATING STATIONS, UNITS NO. 2 & 3

Plant Name:	San Onofre Nuclear Generating Stations, Units No. 2 & No. 3
Licensing Stage:	OL
Docket Numbers:	50-361/362
Milestone Number:	01-02
Responsible Branch and Project Manager:	LWR 2 H. Rood
Requested Completion Date:	December 22, 1976
Review Status:	Complete

As a result of our review of those portions of the applicant's FSAR for which the Auxiliary and Power Conversion Systems Branch has primary responsibility, we conclude that the FSAR is 90 percent complete and recommend that it be accepted. Our review was based on guidelines provided by Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2. Our review also covered open items identified in the ACRS CP letter to verify that the applicant has addressed them in the FSAR.

The enclosure identifies areas where we need additional information. These areas include flood protection against ground water, missile protection, protection against postulated rupture of piping, spent fuel pool cooling system, fuel handling system, salt water system, component cooling water system, compressed air system, diesel generator supporting systems, circulating water system, feedwater system and auxiliary feedwater system. We have also stated positions with respect to the high and moderate energy piping failure analyses, the spent fuel pool cooling system, and the power source diversity requirements of the auxiliary feedwater system.

Contact:
C. Liang
Ext. 27763

50-362
M 4

DEC 2 2 1976

The applicant, by letter dated October 28, 1976, has indicated an October, 1977, submittal date for the Fire Protection re-evaluation date. This date is unacceptable if we are to complete the review prior to fuel loading. We will inform the applicant that his fire protection re-evaluation has to be submitted prior to June, 1977, in order to complete our safety review in a timely manner.

Victor Benaroya, Chief
Auxiliary and Power
Conversion Systems Branch
Division of Systems Safety

Enclosure:
As stated

cc: S. Hanauer
R. Heineman
R. Boyd
W. McDonald
H. Rood
V. Benaroya
D. Fischer
P. Matthews
R. Tedesco
J. Glynn
D. Vassallo
C. Liang

Auxiliary and Power Conversion Systems Branch
Acceptance Review
San Onofre Nuclear Generating Stations
Units No. 2 & No. 3
Docket Nos. 50-361/362

010.1 (General) Some symbols used on your P&IDs are not defined in Section 1.1 of the FSAR. Revise Figures 1.1-2 through 1.1-5 to include definitions of all the sumbols used.

010.2 (3.4) Provide the results of an analyses to demonstrate that the safety-related systems are protected from flooding due to ground water seepage through seismic Category I building walls.

010.3 (3.5) Provide a tabulation of all safety-related components which are located outdoors and describe the protection to be afforded to these components to prevent their being damaged by tornado generated missiles or a seismic event. Include in this tabulation all HVAC system air intakes and exhaust and the diesel generator combustion air intake and exhaust. Identify the locations of these components, air intakes and exhausts on the plant arrangement drawings.

010.4 (RSP) (3.6) It is our position that San Onofre Units No. 2 & 3 meet the guidelines set forth in the attached Branch Technical Position APCSB 3-1 or follow the guidance provided in the December, 1972 letter from A. Giambusso, Appendix B to the Branch Technical Position APCSB 3-1 plus an analyses made in conformance with B.3 of this position be presented to demonstrate that acceptable

010.4 (RSP) (Cont'd) protection against the effects of piping failures outside containment has been provided.

010.5 (3.6) Provide layout drawings of the safety-related areas outside containment showing the high energy piping systems and their relation to the safety-related equipment. Indicate the method of protection provided against a high energy piping system failure for each system listed in Table 3.6-2; also provide a table listing moderate energy piping systems and their method of protection that resulted from the analyses performed in request 010.4.

010.6 (3.6) In Section 3.6A you state that the results of the high energy pipe break analysis will be submitted as an amendment to this FSAR by approximately November, 1978. This is not consistent with the present schedule. Wee need the analysis at an early date to meet the schedule to complete our safety evaluation. As an example of your analyses of the effects on the safety-related system from the high and moderate energy piping failure, provide the details of your evaluation on the shutdown cooling system (SCS) to demonstrate that the SCS is protected from the consequences of the fluid system piping failure and maintains its design safety function.

010.7 (9.0) Identify the seismic design and quality group classification of the piping systems and the points of change classification in the system for all P&IDs.

010.8 (9.0) In regard to potential failures or malfunctions caused by freezing, icing, and other adverse environmental conditions, discuss the protective measures that are provided to assure the proper function of those components not housed within temperature controlled areas, and that are essential in attaining and maintaining a safe reactor shutdown.

010.9 (RSP) (9.1.3) Discuss the interfaces between the shutdown cooling heat exchangers and the spent fuel pool cooling systems. It is our position that the shutdown cooling system not be used to backup the spent fuel pool cooling system unless the reactor core is unloaded.

010.10 (9.1.4) Provide the results of an analysis to demonstrate that a postulated cask drop will not cause damage of any safety-related system or component which may be located under the travel path of the cask handling. Otherwise, the crane should be designed to meet the guidelines set forth in our Branch Technical Position APCSB 9-1 (attached).

010.11 (9.2.1) Provide detail drawings to show physical arrangement of the salt water system components inside the intake structure and the traveling water screens and screen wash equipment at the intake structure. Provide the seismic Category classification of the traveling water screen systems and discuss the consequences of a failure of the traveling water screen system in light of the service water system operability.

010.12 (9.2.2) In Section 9.2.2 you state that the spent fuel pool heat exchangers are served by the non-seismic Category I portion of the component cooling water system. It is our position that the piping of the spent fuel pool cooling system and the portion of the piping supplying component cooling water to the spent fuel cooling heat exchangers be analyzed for SSE loading and seismic Category I supports to these piping and the components be provided

010.13 (RSP) (9.2.2) The design of your component cooling water system provides a single supply and return line, each supplying cooling water to four reactor coolant pumps. These lines are not designed to seismic Category I requirements and contain motor-operated valves for containment isolation. The seals, and bearings of the reactor coolant pumps require continuous cooling by the component cooling water system during all modes of operation. Inadvertent closure of any one of the above motor-operated valves would terminate the coolant flow to all of the pumps which potentially

010.13 (RSP) (Cont'd)

may lead to fuel damage, due to a locked rotor. Therefore, it is our position that you design this portion of the component cooling water system so that the following criteria are met:

1. A single failure in the component cooling water system shall not result in fuel damage or damage to the reactor coolant system pressure boundary caused by an extended loss of cooling to the reactor coolant pumps. Single failure includes operator error, spurious actuation of motor-operated valves, and loss of component cooling water pumps.
2. A moderate energy leakage crack or an accident that is initiated from a failure in the component cooling water system piping shall not result in excessive fuel damage or a breech of the reactor coolant system pressure boundary when an extended loss of cooling to the reactor coolant pumps occurs. A single active failure shall be considered when evaluating the consequences of the accident. Moderate leakage cracks should be determined in accordance with the guidelines of Branch Technical Position APCSB 3-1, "Protection Against Postulated Failures in Fluid Systems Outside Containment."

To meet the two criteria above, that portion of the component cooling water system which supplies cooling water to the reactor coolant pumps can be designed to non-seismic Category I requirements and Quality Group D if you demonstrate that the reactor coolant pumps are capable to operate with loss of cooling for

010.13 (RSP) (Cont'd)

longer than 30 minutes without loss of function and the need for operator protective action. And, safety grade instrumentation to detect the loss of component cooling water to the reactor coolant pumps and to alarm the operator in the control room is provided. The entire instrumentation system, including audible and visual status indicators for loss of component cooling water should meet the requirements of IEEE Std 279-1971. Alternately, if it cannot be demonstrated that the reactor coolant pumps will operate longer than 30 minutes without loss of function or operator corrective action, then your design must meet the following requirements for the entire component cooling water system:

1. safety grade instrumentation consistent with the criteria for the protection system shall be provided to initiate automatic protection of the plant. For this case, the component cooling water supply to the seal and bearing of the pumps may be designed to non-seismic Category I requirements and Quality Group D; or
2. the component cooling water supply to the pumps shall be capable of withstanding a single active failure or a moderate energy line crack as defined in our Branch Technical Position APCSB 3-1 and be designed to seismic Category I, Quality Group C and ASME Section III, Class 3 requirements.

010.14 (RSP) (9.3.1)

You indicate in Section 9.3.1 that the compressed air system is not designed to seismic Category I requirements and the

010.14 (RSP) (Cont'd) atmospheric steam dump valves are designed to fail-shut in the event of loss of air supply. It is our position that the atmospheric steam dump valves should be able to be operated from the control room for cold shutdown of the plant. Also, a seismic Category I air supply to the steam dump valves should be provided.

010.15 (9.5.1) The description of the fire protection system provided in your FSAR does not provide all of the information requested by Regulatory Guide 1.70.4. In addition, your design should meet the guidelines of Appendix A to Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." Revise you design as necessary to meet these guidelines.

010.16 (9.5) Provide a detailed description, safety evaluation, and P&IDs of the diesel generator supporting systems (i.e., diesel generator fuel oil storage and transfer system, starting system, diesel generator lubrication system and diesel generator combustion air intake and exhaust system).

010.17 (10.4.5) Provide the evaluation regarding the effects of possible circulating water system failure inside the turbine building. Include the following:

1. The maximum flow rate through a complete failed expansion joint.

010.17 (Cont'd)

2. The potential for and the means provided to detect a failure in the circulating water transport system barrier such as the rubber expansion joints. Include the design and operating pressures of the various portions of the transport system barrier and their relation to the pressures which could exist during malfunctions and failures in the system (rapid valve closure).

3. The time required to stop the circulating water flow (time zero being the instant failure) including all inherent delays such as operator reaction time, drop out times of the control circuitry and coastdown time.

4. For each postulated failure in the circulating water transport system barrier give the rate of rise of water in the associated spaces and total height of the water when the circulating water flow has not been stopped or overflows to site grade.

5. For each flooded space provde a discussion, with the aid of drawings, of the protective barrier provided for all essential systems that could become affected as a result of flooding. Include a discussion of the consideration given to passageways, pipe chases, and/or the cableways joining the flooded space to the spaces containing safety-related system components. Discuss the effect of the flood water on all submerged essential electrical systems and components.

010.18 (10.4.7)

Events such as damage to the feedwater system piping at Indian Point Unit No. 2, November 13, 1973, and at other plants, could originate as a consequence of uncovering of the feedwater sparger in the steam generator or uncovering of the steam generator feedwater inlet nozzles. Subsequent events in turn lead to the generation of a pressure wave that is propagated through the pipes and could result in unacceptable damage.

1. Describe normal operating occurrences of transients that could cause the water level in the steam generator to drop below the sparger or nozzles to cause uncovering and allow steam to enter the sparger and feedwater piping.
2. Describe your criteria or show by isometric diagrams, the routing of the feedwater piping from the steam generators outwards to beyond containment structure up to the outer isolation valve and restraint.
3. Describe any analysis on the piping system including any forcing functions that will be performed or the results of test programs to verify that either, uncovering of feedwater lines could not occur or that if it did occur, unacceptable damages such as the experience at the Indian Point Unit No. 2 facility would not result at your facility.

010.19 (RSP)
(10.4.9)

It is our position that the power sources for all controls, valve operators and other supporting systems (e.g., pump lube oil cooling system) associated with the trubine driven auxiliary feedwater pump be independent from A/C power. This is to comply with the diversity requirement in attached Branch Technical Position APCSB 10-1. Modify the system design to comply with this position and confirm that the turbine driven pump lube oil cooler will receive cooling water from the pump recirculation line.

January 1975

BRANCH TECHNICAL POSITION APCSB 3-1

PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT

A. BACKGROUND

General Design Criterion 4, "Environmental and Missile Design Bases," of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," requires that systems and components important to safety" ... shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit." Guidance on acceptable design approaches to meet General Design Criterion 4 for exisitng plants and for plants for which applications for construction permits were then under review was provided in letters to applicants and licensees from A. Giambusso, Deputy Director of Licensing for Reactor Projects, most of which were dated in December 1972. The guidance document from these letters is attached as Appendix B to this position. Similar interim guidance for new plants was provided in a letter to applicants, prospective applicants, reactor vendors, and architect-engineers from J. F. O'Leary, Director of Licensing, dated July 12, 1973. This document is attached as Appendix C to this position.

Guidance is available for protection against pipe whipping and other effects of postulated fluid system piping failures (e.g., a break or rupture resulting in a loss-of-coolant accident) of systems and components important to safety located within primary reactor containment. As an example, this problem is addressed by Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

Reviews of nuclear power plant designs have indicated that the functional or structural integrity of systems and components required for safe shutdown of the reactor and maintenance of the cold shutdown conditions could be endangered by fluid system piping failures at locations outside containment. The staff has evolved an acceptable approach for the design including the arrangement, of fluid systems located outside of containment. This approach is set forth in this position and in the companion branch technical position (BTP) attached to Standard Review Plan 3.6.2, BTP MEB 3-1.

It is the intent of this design approach that postulated piping failures in fluid systems should not cause a loss of function of essential safety-related systems and that nuclear plants should be able to withstand postulated failures of any fluid system piping outside containment, taking into account the direct results of such a failure and the further failure of any single active component, with acceptable offsite consequences.

The detailed provisions of the position below and of BTP MEB 3-1 are intended to implement this intent with due consideration of the special nature of certain dual purpose systems and the need to define and to limit to a finite number the types and locations of piping failures to be analyzed. Although various measures for the protection of safety-related systems and components are outlined in this position, the preferred method of protection is based upon separation and isolation by plant arrangement.

B. BRANCH TECHNICAL POSITION

1. Plant Arrangement

Protection of essential systems and components[1/] against postulated piping failures in high or moderate energy fluid systems that operate during normal plant conditions and that are located outside of containment should be provided by one of the following plant arrangement considerations:

a. Plant arrangements should separate fluid system piping from essential systems and components. Separation should be achieved by plant physical layouts that provide sufficient distances between essential systems and components and fluid system piping such that the effects of any postulated piping failure therein (e.g., pipe whip, jet impingement, and the environmental conditions resulting from the escape of contained fluids as appropriate to high or moderate-energy fluid system piping) cannot impair the integrity or operability of essential systems and components.

b. Fluid system piping or portions thereof not satisfying the provisions of B.1.a should be enclosed within structures or compartments designed to protect nearby essential systems and components. Alternatively, essential systems and components may be enclosed within structures or components designed to withstand the effects of postulated piping failures in nearby fluid systems.

c. Plant arrangements or system features that do not satisfy the provisions of either B.1.a or B.1.b should be limited to those for which the above provisions are impractical because of the stage of design or construction of the plant; because the plant design is based upon that of an earlier plant accepted by staff as a base plant under the Commission's standardization and replication policy or for other substantive reasons

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1/ See Appendix A for definitions of underlined phrases.

such as particular design features of the fluid systems. Such cases may arise, for example, (1) at interconnections between fluid systems and essential systems and components, or (2) in fluid systems having dual functions (i.e., required to operate during normal plant conditions as well as to shut down the reactor). In these cases, redundant design features that are separated or otherwise protected from postulated piping failures, or additional protection, should be provided so that the effects of postulated piping failures are shown by the analyses and guidelines of B.3 to be acceptable. Additional protection may be provided by restraints and barriers or by designing or testing essential systems and components to withstand the effects associated with postulated piping failures.

2. Design Features

a. Essential systems and components should be designed to meet the seismic design requirements of Regulatory Guide 1.29.

b. Protective structures or compartments, fluid system piping restraints, and other protective measures should be designed in accordance with the following:

(1) Protective structures or compartments needed to implement B.1 should be designed to seismic Category I requirements. The protective structures should be designed to withstand the effects of a postulated piping failure (i.e., pipe whip, jet impingement, pressurization of compartments, water spray, and flooding, as appropriate) in combination with loadings associated with the operating basis earthquake and safe shutdown earthquake within the respective design load limits for structures. Piping restraints, if used, may be taken into account to limit effects of the postulated piping failure.

(2) High-energy fluid system piping restraints and protective measures should be designed such that a postulated break in one pipe cannot, in turn, lead to rupture of other nearby pipes or components if the secondary rupture could result in consequences that would be considered unacceptable for the initial postulated break. An unrestrained whipping pipe should be considered capable of (a) rupturing impacted pipes of smaller nominal pipe sizes and (b) developing through-wall leakage cracks in larger nominal pipe sizes with thinner wall thicknesses, except where experimental or analytical data for the expected range of impact energies demonstrate the capability to withstand the impact without failure.

c. Fluid system piping between containment isolation valves should meet the following design provisions:

(1) Portions of fluid system piping between isolation valves or single barrier containment structures (including any rigid connection to the containment penetration) that connect, on a continuous or intermittent bases, to the reactor coolant pressure boundary, or the steam and feedwater systems of PWR plants, should be designed to the stress limits specified in B.1.b or B.2.b of Branch Technical Position (BTP) MEB 3-1, attached to Standard Review Plan 3.6.2.

These portions of high-energy fluid system piping should be provided with pipe whip restraints that are capable of resisting bending and torsional moments produced by a postulated piping failure either upstream or downstream of the containment isolation valves. The restraints should be located reasonably close to the containment isolation valves and should be designed to withstand the loadings resulting from a postulated piping failure beyond these portions of piping so that neither isolation valve operability nor the leaktight integrity of the containment will be impaired.

(2) Portions of fluid system piping between isolation valves of dual barrier containment structures should also meet the design provisions of B.2.c.(1). In addition, those portions of piping that pass through the containment annulus, and whose postulated failure could affect the leaktight integrity of the containment structure or result in pressurization of the containment annulus beyond design limits should be provided with an enclosing protective structure.

For the purpose of establishing the design parameters (i.e., pressure, temperature) of the enclosing protective structure, a full flow area opening should be assumed in that portion of piping within the enclosing structure and taking into account vent areas, if provided, in the enclosing structure. Where guard pipes for individual process pipes are used as an enclosing protective structure, such guard pipes should be designed to meet the requirements specified in B.1.b.(6) of BTP MEB 3-1.

(3) Terminal ends of the piping runs extending beyond these portions of high-energy fluid system piping should be considered to originate at a point immediately outside or beyond these required pipe whip restraints located inside and outside containment.

d. Inservice examination and related design provisions should be in accordance with the following:

(1) The protective measures, structures, and guard pipes should not prevent the access required to conduct the inservice examinations specified in the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, "Rules for Inspection and Testing of Components in Light-Water Cooled Plants."

(2) For those portions of high energy fluid system piping identified in B.2.c, the extent of inservice examinations completed during each inspection interval (IWA-2400 ASME Code, Section XI) should provide 100 percent volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping.

(3) For those portions of fluid systems piping enclosed in guard pipes, inspection ports should be provided in guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in that portion of the guard pipe passing through the annulus of dual barrier containment structures.

(4) The areas subject to examination should be defined in accordance with Examination Categories C-F and C-G for Class 2 piping welds in Table IWC-2520.

3. Analyses and Effects of Postulated Piping Failures

a. To show that the plant arrangement and design features provide the necessary protection of essential systems and components, piping failures should be postulated in accordance with BTP MEB 3-1, attached to Standard Review Plan 3.6.2. In applying the provisions of BTP MEB 3-1, each longitudinal or circumferential break in high-energy fluid system piping or leakage crack in moderate energy fluid system piping should be considered separately as a single postulated initial event occurring during normal plant conditions. An analysis should be made of the effects of each such event, taking into account the provisions of BTP MEB 3-1 and of the system and component operability considerations of B.3.b below. The effects of each postulated piping failure should be shown to result in offsite consequences within the guidelines of 10 CFR Part 100 and to meet the provisions of B.3.c and d below.

b. In analyzing the effects of postulated piping failures, the following assumptions should be made with regard to the operability of systems and components:

(1) Offsite power should be assumed to be unavailable if a trip of the turbine-generator system or reactor protection system is a direct consequence of the postulated piping failure.

(2) A single active component failure should be assumed in systems used to mitigate consequences of the postulated piping failure and to shut down the reactor, except as noted in B.3.b.(3) below. The single active component failure is assumed to occur in addition to the postulated piping failure and any other direct consequences of the piping failure, such as unit trip and loss of offsite power.

(3) Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential system, i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure, single failures of components in the other train or trains of that system only need not be assumed provided the system is designed to seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems. Examples of systems that may, in some plant designs, qualify as dual-purpose essential systems include service water systems, component cooling systems, and residual heat removal systems.

(4) All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of a postulated piping failure. In judging the availability of systems, account should be taken of the postulated failure and its direct consequences such as unit trip and loss of offsite power, and of the assumed single active component failure and its direct consequences. The feasibility of carrying out operator actions should be judged on the basis of ample time and adequate access to equipment being available for the proposed actions.

c. The effects of a postulated piping failure, including environmental conditions resulting from the escape of contained fluids, should not preclude habitability of the control room or access to surrounding areas important to the safe control of reactor operations needed to cope with the consequences of the piping failure.

d. A postulated failure of piping not designed to seismic Category I standards should not result in any loss of capability of essential systems and components to withstand the further effects of any single active component failure and still perform all functions required to shut down the reactor and mitigate the consequences of the postulated piping failure.

4. Implementation

a. Designs for plants for which construction permit applications are tendered after July 1, 1975 should conform to the provisions of this position.

b. Design of plants for which construction permit applications are tendered after July 1, 1973 and before July 1, 1975 should conform to the provisions of either (a) the letter of July 12, 1973 from J. F. O'Leary, Appendix C to this position, or (b) this position, at the option of the applicants.

c. Design of plants for which construction permit applications were tendered before July 1, 1973 and for which operating licenses are issued after July 1, 1975 should follow the guidance provided in the December 1972 letter from A. Giambusso Appendix B to this position. Analyses, made in conformance with B.3 of this position, should be presented as part of the operating license application for these plants to demonstrate that acceptable protection against the effects of piping failures outside containment has been provided. Alternately, this position may be used in its entirety as an acceptable basis for this finding.

For plants in this category for which construction permits are not issued as of February 1, 1975, a commitment by the applicant to either (a) follow the guidance of Appendix B and submit B.3 analyses with the plant final safety analysis report (FSAR), or (b) conform the plant design to the provisions of this position, should provide an acceptable basis for issuance of the construction permit with regard to effects of piping failures outside containment.

d. Designs of plants for which operating licenses are issued before July 1, 1975 are considered acceptable with regard to effects of piping failures outside containment on the basis of the analyses

made and measures taken by applicants and licensees in responses to the December 1972 letter from A. Giambusso, Deputy Director for Reactor Projects, Directorate of Licensing, and the staff review and acceptance of these analyses and measures.

For plants in this category for which the staff review and acceptance of protection against the effects of piping failures outside containment is not substantially complete as of February 1, 1975 a commitment by the applicant to carry out analyses according to B.3 of this position, to submit them for staff review, and to carry out any system modifications found necessary before extended operation of the plant at power levels above one-half the license power level, should provide an acceptable basis for issuance of the operating license.

C. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

3. Letter from A. Giambusso, Deputy Director for Reactor Projects, Directorate of Licensing, to applicants and licensees, December 1972, and attachment entitled "General Information Required for Consideration of the Effects of a Piping System Break Outstide Containment." The corrected attachment is Appendix B to this position.

4. Letter from J. O'Leary, Director of Licensing, to applicants, reactor vendors and architect-engineers on the subject of postulated piping failures outside containment dated July 12, 1973. The letter with attachment is Appendix C to this position.

5. ASME Boiler and Pressure Vessel Code, Sections III and XI, American Society of Mechanical Engineers.

APPENDIX A

Branch Technical Position APCSB 3-1

DEFINITIONS

Essential Systems and Components. Systems and components required to shut down the reactor and mitigate the consequences of a postulated piping failure, without offsite power.

Fluid Systems. High and moderate energy fluid systems that are subject to the postulation of piping failures outside containment against which protection of essential systems and components is needed.

High-Energy Fluid Systems. Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either or both of the following are met:

a. Maximum operating temperature exceeds 200°F, or

b. Maximum operating pressure exceeds 275 psig.

Moderate-Energy Fluid Systems. Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

a. Maximum operating temperature is 200°F or less, and

b. Maximum operating pressure is 275 psig or less.

Normal Plant Conditions. Plant operating conditions during reactor startup, operation at power hot standby, or reactor cooldown to cold shutdown condition.

Upset Plant Conditions. Plant operating conditions during system transients that may occur with moderate frequency during plant service life and are anticipated operational occurrences, but not during system testing.

Postulated Piping Failures. Longitudinal and circumferential breaks in high-energy fluid system piping and through-wall leakage cracks in moderate-energy fluid system piping postulated according to the provisions of BTP - MEB 3-1.

S_h and S_A. Allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME Code, Section III.

S_m. Design stress intensity as defined in Article NB-3600 of the ASME Code, Section III.

Single Active Component Failure. Malfunction or loss of function of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, pneumatic or electrical malfunction, but not the loss of component structural integrity. The direct consequences of a single active component failure are considered to be part of the single failure.

Terminal Ends. Extremities of piping runs that connect to structures, components (e.g., vessel pumps, valves), or pipe anchors that act as rigid constraints to piping thermal expansion. A branch connection to a main piping run is a terminal end of the branch run.

In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve) a terminal end of such runs is the piping connection to this closed valve.

APPENDIX B

BRANCH TECHNICAL POSITION APCSB 3-1

This appendix consists of the attachment to the letters sent by A. Giambusso, Deputy Director for Reactor Projects, Directorate of Licensing, in December 1972 to applicants and licensees on the subject of postulated piping failures outside containment. The attachment provided guidance on measures to be taken and on information to be submitted. An errata sheet for the attachment was sent in January 1973 to recipients of the original letters. The attachment as given here has been corrected to include the errata.

General Information Required for Consideration of the Effects of a Piping System Break Outside Containment

The following is a general list of information required for AEC review of the effects of a piping system break outside containment, including the double-ended rupture of the largest pipe in the main steam and feedwater systems, and for AEC review of any proposed design changes that may be found necessary. Since piping layouts are substantially different from plant to plant, applicants and licensees should determine on an individual plant basis the applicability of each of the following items for inclusion in their submittals.

1. The systems (or portions of systems) for which protection against pipe whip is required should be identified. Protection from pipe whip need not be provided if any of the following conditions will exist:

   (a) Both of the following piping system conditions are met:

      (1) the service temperature is less than 200°F; and

      (2) the design pressure is 275 psig or less; or

   (b) The piping is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, or restrained from whipping by plant design features, such as concrete encasement; or

(c) Following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any structure, system or component important to safety; or

(d) The internal energy level[1/] associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system or component to an unacceptable level.

2. Design basis break locations should be selected in accordance with the following pipe whip protection criteria: however, where pipes carrying high enery fluids are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, supplemental protection of those structures and systems shall be provided to cope with the environmental effects (including the effects of jet impingement) of a single postulated open crack at the most adverse location(s) with regard to those essential structures and systems, the length of the crack being chosen not to exceed the critical crack size. The critical crack size is taken to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width.

The criteria used to determine the design basis piping break locations in the piping systems should be equivalent to the following:

(a) ASME Section III Code Class I piping[2/] breaks should be postulated to occur at the following locations in each piping run[3/] or branch run:

(1) The terminal ends;

(2) Any intermediate locations between terminal ends where the primary plus secondary stress intensities S_n (circumferential or longitudinal) derived on an elastically calculated basis under the loadings associated with one-half safe shutdown earthquake and operational plant conditions[4/] exceeds 2.0 S_m[5/] for ferritic steel, and 2.4 S_m for austenitic steel;

(3) Any intermediate locations between terminal ends where the cumulative usage factor (U)[6/] derived from the piping fatigue analysis and based on all normal, upset, and testing plant conditions exceeds 0.1; and

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[1]Footnotes are collected at the end of this appendix.

(4) At intermediate locations in addition to those determined by (1) and (2) above, selected on a reasonable basis as necessary to provide protection. At a minimum, there should be two intermediate locations for each piping run or branch run.

(b) ASME Section III Code Class 2 and 3 piping breaks should be postulated to occur at the following locations in each piping run or branch run:

(1) The terminal ends;

(2) Any intermediate locations between terminal ends where either circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.8 $(S_h + S_A)$[7/] or the expansion stresses exceed 0.8 S_A; and

(3) Intermediate locations in addition to these determined by (2) above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

3. The criteria used to determine the pipe break orientation at the break locations as specified under (2) above should be equivalent to the following:

(a) Longitudinal[8/] breaks in piping runs and branch runs, 4 inches nominal pipe size and larger, and/or

(b) Circumferential[9/] breaks in piping runs and branch runs exceeding 1 inch nominal pipe size.

4. A summary should be provided of the dynamic analyses applicable to the design of Category I piping and associated supports which determine the resulting loadings as a result of a postulated pipe break including:

(a) The locations and number of design basis breaks on which the dynamic analyses are based.

(b) The postulated rupture orientation, such as a circumferential and/or longitudinal break(s), for each postulated design basis break location.

(c) A description of the forcing functions used for the pipe whip dynamic analyses including the direction, rise time, magnitude, duration, and initial conditions that adequately represent the jet stream dynamics and the system pressure difference.

(d) Diagrams of mathematical models used for the dynamic analysis.

(e) A summary of the analyses which demonstrates that unrestrained motion of ruptured lines will not damage to an unacceptable degree, structure, systems, or components important to safety, such as the control room.

5. A description should be provided of the measures, as applicable, to protect against pipe whip, blowdown jet, and reactive forces including:

   (a) Pipe restraint design to prevent whip impact;

   (b) Protective provisions for structures, systems, and components required for safety against pipe whip and blowdown jet and reactive forces:

   (c) Separation of redundant features;

   (d) Provisions to separate physically piping and other components of redundant features; and

   (e) A description of the typical pipe whip restraints and a summary of number and location of all restraints in each system.

6. The procedures that will be used to evaluate the structural adequacy of Category I structures and to design new seismic Category I structures should be provided including:

   (a) The method of evaluating stresses, e.g., the working stress method and/or the ultimate strength method that will be used;

   (b) The allowable design stresses and/or strains; and

   (c) The load factors and the load combinations.

7. The structural design loads, including the pressure and temperature transients, the dead, live, and equipment loads, and the pipe and equipment static, thermal, and dynamic reactions should be provided.

8. Seismic Category I structural elements such as floors, interior walls, exterior walls, building penetrations, and the buildings as a whole should be analyzed for eventual reversal of loads due to the postulated accident.

9. If new openings are to be provided in existing structures, the capabilities of the modified structures to carry the design loads should be demonstrated.

10. Verification that failure of any structure, including non-seismic Category I structures, caused by the accident, will not cause failure of any other structure in a manner to adversely affect:

(a) Mitigation of the consequences of the accidents; and

(b) Capability to bring the unit(s) to a cold shutdown condition.

11. Verification that rupture of a pipe carrying high energy fluid will not directly or indirectly result in:

(a) Loss of required redundancy in any portion of the protection system (as defined in IEEE-Std 279), Class IE electric system (as defined in IEEE-Std. 308), engineered safety feature equipment, cable penetrations, or their interconnecting cables required to mitigate the consequences of that accident and place the reactor(s) in a cold shutdown condition; or

(b) Environmentally induced failures caused by a leak or rupture of the pipe which would not of itself result in protective action but does disable protection functions. In this regard, a loss of redundancy is permitted; but a loss of function is not permitted. For such situations, plant shutdown is required.

12. Assurance should be provided that the control room will be habitable and its equipment functional after a steam line or feedwater line break or that the capability for shutdown and cooldown on the unit(s) will be available in another habitable area.

13. Environmental qualification should be demonstrated by test for that electrical equipment required to function in the steam-air environment resulting from a high energy line break. The information required for our review should include the following:

(a) Identification of all electrical equipment necessary to meet requirements of (11) above. The time after the accident in which they are required to operate should be given.

(b) The test conditions and the results of test data showing that the systems will perform their intended function in the environment resulting from the postulated accident and time interval of the accident. Environmental conditions used for the tests should be selected from a conservative evaluation of accident conditions.

(c) The results of a study of steam systems identifying locations where barriers will be required to prevent steam jet impingement from disabling a protection system. The design criteria for the barriers should be stated and the capability of the equipment to survive within the protected environment should be described.

(d) An evaluation of the capability for safety-related electrical equipment in the control room to function in the environment that may exist following a pipe break accident should be provided. Environmental conditions used for the evaluation should be selected from conservative calculations of accident conditions.

(e) An evaluation to assure that the onsite power distribution system and onsite sources (diesels and batteries) will remain operable throughout the event.

14. Design diagrams and drawings of the steam and feedwater lines including branch lines showing the routing from containment to the turbine building should be provided. The drawings should show elevations and include the location relative to the piping runs of safety related equipment including ventilation equipment, intakes, and ducts.

15. A discussion should be provided of the potential for flooding of safety-related equipment in the event of failure of a feedwater line or any other line carrying high energy fluid.

16. A description should be provided of the quality control and inspection programs that will be required or have been utilized for piping systems outside containment.

17. If leak detection equipment is to be used in the proposed modifications, a discussion of its capabilities should be provided.

18. A summary should be provided of the emergency procedures that would be followed after a pipe break accident, including the automatic and manual operations required to place the reactor unit(s) a a cold shutdown condition. The estimated times following the accident for all equipment and personnel operational actions should be included in the procedure summary.

19. A description should be provided of the seismic and quality classification of the high energy fluid piping systems including the steam and feedwater piping that run near structures, systems, or components important to safety.

20. A description should be provided of the assumptions, methods, and results of analyses, including steam generator blowdown, used to calculate the pressure and temperature transients in compartments, pipe tunnels, intermediate buildings, and the turbine building following a pipe rupture in these areas. The equipment assumed to function in the analyses should be identified and the capability of systems required to function to meet a single active component failure should be described.

21. A description should be provided of the methods or analyses performed to demonstrate that there will be no adverse effects on the primary and/or secondary containment structures due to a pipe rupture outside these structures.

1/ The internal fluid energy level associated with the pipe break reaction may take into account any line restrictions (e.g., flow limiter) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions, as applicable. The energy level in a whipping pipe may be considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

2/ Piping is a pressure-retaining component consisting of straight or curved pipe and pipe fittings (e.g., elbows, tees, and reducers).

3/ A piping run interconnects components such as pressure vessels, pumps, and rigidly fixed valves that may act to restrain pipe movement beyond that required for design thermal displacement. A branch run differs from a piping run only in that it originates at a piping intersection, as a branch of the main pipe run.

4/ Operational plant conditions include normal reactor operation, upset conditions (e.g., anticipated operational occurrences) and testing conditions.

5/ S_m is the design stress intensity as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Plant Components."

6/ U is the cumulative usage factor as specified in Section III of the AMSE Boiler and Pressure Vessel Code, "Nuclear Power Plant Components."

7/ S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code Section III Winter 1972 Addenda.

S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967.

8/ Longitudinal breaks are parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the effective cross-sectional flow area upstream of the break location. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.

9/ Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially, and cause whipping in any direction normal to the pipe axis.

APPENDIX C

BRANCH TECHNICAL POSITION APCSB 3-1

This appendix consists of the letter and attachment sent by J. F. O'Leary, Director of Licensing, to applicants, reactor vendors, and architect-engineers on the subject of postulated piping failures outside containment. The letter was dated July 12, 1973.

Late last year, the Atomic Energy Commission's Regulatory staff requested those utilities that operate nuclear power plants, have applied for operating licenses, or have plants whose construction permit review was essentially complete, to assess the effects and consequences of a postulated rupture of piping containing high-energy fluids and located outside of the containment structure. These requests were issued by Mr. A. Giambusso, Deputy Director of Reactor Projects, Directorate of Licensing, in letters, most of which were dated in December 1972.

Because these plants were either in operation or in advanced stages of engineering design and construction, the request included guidance for corrective modifications that could be implemented by in-situ measures. Such modifications included relocation or rerouting of piping, installation of impingement barriers and encapsulation sleeves around high stressed piping regions, provisions for venting of compartments subject to pressurization, addition of piping restraints, and strengthening of structural components of buildings.

From our review of responses submitted to the Regulatory staff, and from discussions with architect-engineering firms, we have learned that some of these organizations have inferred that the criteria contained in Mr. A. Giambusso's letter pertaining to corrective modifications for plants in advanced stages of construction and operation are applicable for the design of high-energy fluid systems outside the containment in new designs of nuclear power plants. It was not our intent that the criteria for corrective plant modifications be applied to new power plants that are in the initial design stages. We believe that a more direct approach, involving a rearrangement of the physical plant layout with a view to relocation of essential safety systems and components is appropriate for the new plants.

For the present, pending issuance of a planned AEC Regulatory Guide - "Protection Against Postulated Events and Accidents Outside Containment," an acceptable implementation of Criterion 4 of the Commission's General Design Criteria listed in Appendix A of 10 CFR Part 50, as applied to new plants with respect to the design of structures, systems, and components important to safety and located outside of containment is as follows:

I. PIPING SYSTEMS CONTAINING HIGH-ENERGY FLUIDS* DURING NORMAL REACTOR OPERATION

(a) The piping systems are isolated by adequate physical separation and remotely located from safety systems and components that are required to shut down the reactor safely and maintain the plant in a cold shutdown condition.

(b) Where isolation by remote location is impracticable, systems containing high-energy fluids, or portions of the systems, are enclosed within the structures suitably designed to protect adjoining safety systems and components required to shut down the reactor safely and maintain the plant in a cold shutdown condition from postulated pipe failures within the enclosure.

(c) Where both isolation by remote location (as specified in I.a) and enclosure in protective structures (as specified in I.b) are impracticable, systems containing high-energy fluids, or portions of the systems, are provided with restraints and protective measures such that the operability and integrity of structures, safety systems and components that are required to shut down the reactor safely and maintain the plant in a cold shutdown condition are not impaired.

(d) Protective enclosures for the piping systems containing high-energy fluids are designed as Seismic Category I structures to withstand the combined effects of a postulated pipe break, the dynamic effects of pipe shipping, the jet impingement forces, and the compartment pressurization as a consequence of discharging fluids in combination with the specified seismic event of the Safe Shutdown Earthquake and normal operating loads.

(e) Piping systems containing high-energy fluids are designed so that the effects of a single postulated pipe break cannot, in turn, cause failures of other pipes or components with unacceptable consequences.

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*Refer to Appendix A for identification of high-energy fluid systems.

In addition, any systems, or portions of systems that are designed to mitigate the consequences of a postulated pipe failure, and to place the reactor in the cold shutdown condition, are provided with design features that will assure the performance of their safety function, assuming a single active component failure.

(f) For a postulated pipe failure, the escape of steam, water, and heat from structures enclosing the high-energy fluid containing piping does not preclude: 1) the accessibility to surrounding areas important to the safe control of reactor operations, 2) the habitability of the control room, 3) the ability of instrumentation, electric power supplies, and components and controls to initiate, actuate and complete a safety action. In this regard, a loss of redundancy is permissible but not the loss of function.

(g) The criteria for determination of postulated break locations are contained in the attached Appendix A, "Criteria for Determination of Postulated Pipe Break or Leakage Locations in Fluid Piping Systems Outside Containments."

II. PIPING SYSTEMS CONTAINING MODERATE-ENERGY FLUIDS* DURING REACTOR OPERATION

(a) Piping systems containing moderate-energy fluids are designed to comply with the criteria applied to high-energy fluid piping systems as listed under I., above, except that the piping is postulated to develop a limited-size through-wall leakage crack instead of a pipe break.

(b) For each postulated leakage, design measures are included that provide protection from the effects of the resulting water spray and flooding to the same extent required to satisfy criterion I(e).

(c) The criteria for determination of postulated leakage locations are contained in Appendix A.

The measures taken for the protection of structures, systems and components important to safety should not preclude the conduct of inservice examinations of ASME Class 2 and 3 pressure-retaining components as required by the rules of ASME Boiler and Pressure Vessel Code - Section XI, "Inservice Inspection of Nuclear Power Plant Components."

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*Refer to Appendix A for identification of moderate-energy fluid systems.

Although compliance with the design criteria listed above should be accomplished by plant arrangement and layouts utilizing the separation concept to the extent practicable, special consideration will be necessary to provide adequate protection where interconnection is unavoidable between high-energy fluid containing piping and piping of systems important to safety.

We are prepared to discuss with you these guidelines for the design of new nuclear power plants with regard to protection required against postulated breaks of high and moderate energy piping outside of containment, particularly for those plants with construction permit applications currently under consideration.

Sincerely,

John F. O'Leary, Director
Directorate of Licensing

Enclosure:
Appendix A

APPENDIX A

(To J. F. O'Leary Letter Dated July 12, 1973)

CRITERIA FOR DETERMINATION OF POSTULATED BREAK AND LEAKAGE LOCATIONS IN HIGH[1/] AND MODERATE[2/] ENERGY FLUID PIPING SYSTEMS OUTSIDE OF CONTAINMENT STRUCTURES[10/]

A. High-Energy Fluid Systems

1. For piping systems that by plant arrangement and layout are isolated by remote location from structures, systems, and components important to safety[3/], pipe breaks[4/] need not be postulated provided the requirements of A.4 are satisfied.

2. For piping systems that are enclosed in suitably designed concrete structures or compartments to protect structures, systems, and components important to safety, pipe breaks should be postulated at the following locations in each piping or branch run within the protective structure:

   a. the terminal ends[9/] of the piping or branch run (except as exempted by the provisions of A.4), if located within the protective structure or compartment, and

   b. each fitting (i.e., elbow, tee, cross, non-standard fitting), and

   c. a minimum of one break selected in each piping or branch run within the protective structure or compartment at a location that results in the maximum loading from the impact of the postulated ruptured pipe and jet discharge force on wall, floor, and roof of the structure or compartment, including internal pressurization, and taking into account any piping restraints provided to limit pipe motions.

3. For portions of piping systems that can neither be isolated as specified in A.1, nor enclosed in protective structures as specified in A.2, pipe breaks should be postulated at the following locations in each piping or branch run within the confines of the structures or compartments that enclose or adjoin areas containing systems and components important to safety:

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[1]Footnotes are collected at the end of this Appendix.

a. the terminal ends[9] of piping or branch run (except as exempted by A.4), if located within the boundary of the confining structure or each compartment within the structure; and

b. any intermediate location within the boundary of the confining structure or each compartment within the structure where the stresses[5] under the loadings associated with specified seismic events[6] and operational plant conditions[7] exceed 0.8 $(S_h + S_A)$[8] or, in lieu of these calculated stress-related locations, at each fitting (i.e., elbow, tee, cross, non-standard fitting); and

c. a minimum of two separated locations within the boundary of the confining structure or each compartment within the structure in piping or branch runs exceeding twenty pipe diameters in length; a minimum of one location in piping or branch runs twenty pipe-diameters or less in length except that no intermediate locations need to be postulated in branch runs that are three pipe-diameters or less in length. Intermediate break locations should be selected such that the maximum pipe whip and jet impingement will result, assuming for this purpose an unrestrained ruptured pipe.

4. For those portions of the piping passing through primary containment penetrations and extending to the first outside isolation valve, pipe breaks need not be postulated provided such piping is conservatively reinforced and restrained beyond the valve such that, in the event of a postulated pipe break outside containment, the transmitted pipe loads will neither impair the operability of the valve nor the integrity of the piping or the containment penetration. (A terminal end of such piping is considered to originate at this restraint location.)

B. Moderate-Energy Fluid Systems

1. For piping systems that by plant arrangement and layout are isolated and physically separated and remotely located from systems and components important to safety, through-wall leakage cracks need not be postulated.

2. For piping systems that are located in the same area as high-energy fluid systems which, by the criteria of A.1 to A.3 have postulated pipe break locations, through-wall leakage cracks need not be postulated.

3. For piping systems that are located in areas containing systems and components important to safety, but where non high-energy fluid systems are present, through-wall leakage cracks should be postulated at the most adverse location to determine the protection needed to withstand the effects of the resulting water spray and flooding.

C. Size and Types of Pipe Breaks and Cracks

1. The following types of breaks should be postulated at the locations specified by the criteria listed under A. High-Energy Fluid Systems:

   a. Longitudinal breaks in piping runs and branch runs with nominal pipe sizes of 4 inches and larger.

   b. circumferential breaks in piping runs and branch runs exceeding a nominal pipe size of 1 inch.

2. The following leakage cracks are postulated at the locations specified by the criteria listed under B. Moderate-Energy Fluid Systems:

   a. through-wall leakage cracks in piping and branch runs exceeding a nominal pipe size of 1 inch, where the crack opening is assumed as 1/2 the pipe diameter in length and 1/2 the pipe wall thickness in width.

FOOTNOTES

1/ High-energy systems include those systems where either of the following conditions are met:

a) the maximum operating temperature exceeds 200°F, and

b) the maximum operating pressure exceeds 275 psig.

2/ Moderate energy systems include those systems where both of the following conditions are met:

a) the maximum operating temperature is 200°F or less, and

b) the maximum operating pressure is 275 psig or less.

3/ Structures, systems, and components important to safety, as specified herein refet to those plant features required to shutdown the reactor safely and maintain the plant in the cold shutdown condition.

4/ Break in piping means (a) a complete circumferential pipe severance and, (b) a longitudinal split opening an area equal to the pipe area, but without pipe severance, Such breaks are assumed to occur at each specified break location, but not concurrently.

5/ Either circumferential or longitudinal stresses derived on an elastically-calculated basis.

6/ Specified seismic events are earthquakes that produce at least 50 percent of the vibratory motion of the Safe Shutdown Earthquake (SSE).

7/ Operational plant conditions include normal reactor operation, upset conditions, (e.g., anticipated operational occurrences) and testing conditions.

8/ S_h is the allowable stress at maximum temperature and S_A is the allowable stress range for expansion stresses, for Class 2 and 3 piping as permitted by the rules of ASME Code Section III.

9/ Terminal ends of pipe runs originate at points of maximum constraint (e.g., Connections to vessels, pumps, valves, fittings that are rigidly anchored to structures) terminal ends of branch runs originate at pipe intersections and components that act as rigid constraints.

10/ These criteria are intended for the purpose of designing piping restraints and do not preclude consideration of other aspects of the AEC General Design Criteria, such as single failure criteria and other additional protective measures required to provide protection against environmental conditions incident to postulated accidents.

April 1975
Revision 0

BRANCH TECHNICAL POSITION APCSB 9-1
OVERHEAD HANDLING SYSTEMS FOR NUCLEAR POWER PLANTS

A. BACKGROUND

Overhead handling systems are used for handling heavy items at nuclear power plants. The handling of heavy loads such as a spent fuel cask raises the possibility of damage to the load and to safety-related equipment or structures under and adjacent to the path on which it is transported should the handling system suffer a breakdown or malfunction.

Two methods are used in nuclear power plants to prevent damage to safety features or release of radioactive material due to dropping of heavy loads, such as a spent fuel cask. One is protection by physical design of the facility to preclude damage to spent fuel and safety-related systems if a heavy load should be dropped. The other is to provide an overhead handling system that is designed so that a connected load would not fall in the event of a failure or malfunction.

An overhead handling system includes all the structural, mechanical, and electrical components that are needed to lift and transfer a load from one location to another. Primary load-bearing components, equipment, and subsystems such as the driving equipment, drum, rope reeving, control, and braking systems require special attention. Proper support of the rope drums ensures that they would be retained and prevented from failing or disengaging from the braking and control system in case of a shaft or bearing failure. If the hoisting system (raising and lowering) includes two mechanical holding brakes, each with better than full-load stopping capacity, that are automatically activated when electric power is off or when mechanically tripped by overspeed or overload devices, a critical load will be safely held or controlled in case of failure in the individual load-bearing parts of the hoisting machinery. Failure of the bridge or trolley travel to stop when power is shut off or an overspeed or overload condition due to malfunction or failure in the drive system can be prevented and controlled by appropriate safety and limit devices and brake systems.

Since the crane industry has not yet developed codes or standards that adequately cover the design, operation, and testing for a "single failure-proof" crane, the APCSB has developed a branch position to provide a consistent basis for reviewing equipment and components for such overhead handling systems. The position below delineates acceptable codes and standards and supplements them with specific recommendations on features that will prevent, control, or stop inadvertent operation or malfunction of the mechanical supporting and moving components of the handling system.

B. BRANCH TECHNICAL POSITION

Overhead handling systems intended to provide single failure-proof handling of loads should be designed so that no single failure or malfunction will result in dropping or loosing control of the heaviest (critical) loads to be handled. Such handling systems should be designed, fabricated, installed, inspected, tested, and operated in accordance with the following:

1. General Performance Specifications

a. Separate performance specifications should be prepared for a permanent crane which is to be used for construction prior to use for plant operation. The allowable design stress limits should be identical for both cases, and the sum total of simultaneously applied loads should not result in stress levels causing any permanent deformation other than that due to localized stress concentrations.

b. The operating environment, including maximum and minimum pressure, temperature, humidity and rates of change of these parameters, should be specified to determine the venting and drainage required for box girder sections. The specifications should also state the corrosive and hazardous conditions that may occur during operation. Fracture toughness for the steel structural materials should be considered. Plate thickness, with a margin for the lowest operating temperatures, should determine the type of steel that can be used with or without toughness tests. The selection of steel materials will be reviewed on a case by case bases.

c. The crane should be classified as seismic Category I and should be capable of retaining the maximum design load during a safe shutdown earthquake, although the crane may not be operable after the seismic event. The bridge and trolley should be provided with means for preventing them from leaving their runways with or without the design load during operation or under seismic loadings. The design rate load plus operational and seismically-induced pendulum and swinging load effects on the crane should be considered in the design of the trolley, and they should be added to the trolley weight for the design of the bridge.

d. All weld joints for load-bearing structures, including those susceptible to lamellar tearing, should be inspected by nondestructive examinations for soundness of the base metal and weld metal.

e. A fatigue analysis should be considered for critical load-bearing structures and components of the crane handling system. The cumulative fatigue usage factors should reflect effects of cyclic loadings from both the construction and operating periods.

f. Preheat and postheat treatment temperatures for all weldments should be specified in the weld procedures. For low-alloy steel, the recommendations of Regulatory Guide 1.50 should be followed.

2. Safety Features

a. The automatic and manual controls and devices required for normal crane operation should be designed such that a malfunction of these controls and devices, and possible subsequent effects during load handling, will not prevent the handling system from being maintained at a safe neutral holding position.

b. Auxiliary systems, dual components, or ancilliary systems should be provided such that in case of subsystem or component failure the load will be retained and held in a stable position.

c. Means should be provided for devices which can be used in repairing, adjusting, replacing failed components or subsystems when failure of an active component or subsystem has occurred and the load is supported and retained in the safe (temporary) position with the system immobile. As an alternative to repairing the crane in place, means may be provided for moving the handling system with load to a laydown area that has been designed for accepting the load and making the repairs.

3. Equipment Selection

a. Dual load attaching points should be provided on the load block or lifting device designed so that each attaching point will be able to support a static load of 3W (W is weight of the design rated load), without permanent deformation other than that due to localized stress concentrations in areas for which additional material has been provided for wear.

b. Lifting devices such as lifting beams, yokes, laddle or trunnion type hooks, slings, toggles, or clevises should be of redundant design with dual or auxiliary devices or combinations thereof. Each device should be designed to support a static load of 3W without permanent deformation.

c. The vertical hoisting (raising and lowering) mechanism which uses rope and consists of upper sheaves (head block), lower sheaves (load block), and rope reeving system, should be designed with redundant means for hoisting. Maximum hoisting speed should be no greater than 5 fpm.

d. The heat and load blocks should be designed to maintain a vertical load balance about the center of lift from the load block through the head block, and should have a dual reeving system. The load block should maintain alignment and a position of stability with either system and be able to support 3W and maintain load stability and vertical alignment from the center of the head block through all hoisting components to the center of gravity of the load.

e. The design of the rope reeving system should be dual, with each system providing separately the load balance on the head and load blocks through the configuration of ropes and rope equalizers. Selection of the hoisting rope or running rope should consider the size, construction, lay, and means or type of lubrication to maintain efficient working of the individual wire strands as the rope passes over the sheaves during the hoisting operation. The effects of impact loadings, acceleration and emergency stops should be included in selection of the rope and reeving system. The wire rope should be 6 x 37 Iron Wire Rope Core (IWRC) or comparable classification.

The stress in the lead line to the drum during hoisting at the maximum design speed with the design rated load should not exceed 20% of the manufacturer's rated strength of the rope. The static stress in rope (load is stationary) should not exceed 12-1/2 of the manufacturer's rated strength. Line speed during hoisting (raising or lowering should not exceed 50 fpm.

f. The maximum fleet angle from drum to lead sheave in the load block should not exceed 3-1/2 degrees at any point during hoisting and there should be only one 180° reverse bend for each rope leaving the drum and reversing on the first or lead sheave on the load block, with no other reverse bends other than at the equalizer if a sheave-type equalizer is used. The fleet angles for rope between individual sheaves should not exceed 1-1/2 degrees. Equalizers may be beam or sheave type. For the recommended 6 x 37 IWRC classification wire rope, pitch diameter of the lead sheave should be 30 times rope diameter for the 180° reverse bend, 26 times rope diameter for running sheaves, and 13 times rope diameter for equalizers. The pitch diameter is measured from the center of the rope in the sheave groove through the sheave center. The dual reeving system may be a single rope from each end of a drum terminating at a beam-type load and rope stretch equalizer with each rope designed for total load, or a 2-rope system may be used from each drum or separate drums with a sheave or beam equalizer, or any other combination which provides two separate and complete reeving systems.

g. The vertical hoisting system components, which include the head block, rope reeving system, load block, and dual load attaching device, should each be designed to sustain a load of 2W (W is the weight of the design rated load). A 2W static load test should be performed for each reeving system and load attaching point at the manufacturer's plant. Each reeving system and each one of the load attaching devices should be assembled with approximately a 6 inch clearance between head and load blocks and should support 200% of the design rated load without degradation of the components or permanent deformation other than that due to localized stress concentrations. Measurements of the geometric configuration of the attaching points should be made before and after test followed by nondestructive examination, which should consist of combination of magnetic particle, ultrasonic, radiographic, and dye penetrant examinations to verify the soundness of fabrication and assure the integrity of this portion of the hoisting system. The results of examinations should be documented and recorded for the hoisting system for each overhead crane.

h. Means should be provided to sense such items as electric current, temperature, overspeed, overloading, and overtravel. Controls should be provided to stop the hoisting movement within 3 inches maximum of vertical travel through a combination of electrical power controls and mechanical braking and torque control systems should one rope of the dual reeving system fail.

i. The control systems may be designed as combination electrical and mechanical systems and may include such items as contractors, relays, resistors, and thyristors in combination with mechanical devices and mechanical braking systems. The electric controls should be selected to provide a maximum breakdown torque limit of 175% of the required rating for a.c. motors or d.c. motors (series or shunt wound) used for the hoisting drive motors. Compound wound d.c. motors should not be used. The control systems provided should consider hoisting (raising and lowering) of all loads, including the design rated load, and the effects of inertia of the rotating hoisting machinery such as motor armatures, shafts and couplings, gear reducers, and drums.

j. The mechanical and structural components of the hoisting system should have the required strength to resist failure should "two-blocking" 1/ or "load hangup" 2/ occur during hoisting. The designer should provide means to absorb or control the kinetic energy of rotating machinery in the event of two-blocking or load hangup. The location and type of mechanical brakes and controls should provide positive and reliable means to stop and hold the hoisting drums for these occurrences. The hoisting system should be able to withstand the maximum torque of the driving motor, if a malfunction occurs and power to the driving motor cannot be shut off at the time of load hangup or two-blocking.

k. The load hoisting drum on the trolley should be provided with structural and mechanic safety devices to prevent the drum from dropping, disengaging from its holding brake system, or rotating, should the drum or any portion of its shaft or bearings fail.

l. To preclude excessive breakdown torque, the horsepower rating (HP) of the electrical motor drive for hoisting should provide no more than 110% of the calculated HP requirement to hoist the design rated load at the maximum design hoist speed.

m. The minimum hoist braking system should include one power control braking system (not mechanical or drag brake-type) and two mechanical holding brakes. The holding brakes should be activated when power is off and should be automatically tripped by mechanical means on overspeed to the full holding position if a malfunction occurs in the electrical brake controls. Each holding brake should be designed to 125% - 150% of maximum developed torque at the point of application (location of the brake in the mechanical drive). The minimum design requirements for braking systems that will be operable for emergency lowering after a single brake failure should be two holding brakes for stopping and controlling drum rotation. Provisions should be made for manual operation of the holding brakes. Emergency brakes or holding brakes which are to be used for manual lowering should be capable of operation with full load and at full travel and provide adequate heat dissipation. Design for manual brake operation during emergency lowering should include features to limit the lowering speed to less than 3.5 fpm.

1/ "Two-blocking" is an inadvertantly continued hoist which brings the load and head block assemblies into physical contact, thereby preventing further movement of the load block and creating shock loads to rope and reeving system.

2/ "Load hangup" occurs when the load block or load is stopped during hoisting by entanglement with fixed objects, thereby overloading the hoisting system.

n. The dynamic and static alignment of all hoisting machinery components including gearing, shafting, couplings, and bearings should be maintained throughout the range of loads be lifted with all components positioned and anchored on the trolley machinery platform.

o. Increment drives for hoisting may be provided by stepless controls or inching motor drives. Plugging 3/ should not be permitted. Controls to prevent plugging should be included in the electrical circuits and the control system. Floating point 4/ in the electrical power system, when required for bridge or tolley movement, should be provided only for the lowest operating speeds.

p. To avoid the possibility of overtorque within the control system, the horsepower rating of the driving motor and gear reducer for trolley and bridge motion of an overhead bridge crane should not exceed 110% of the calculated requirement at maximum speed and with the design rated load. Incremental or fractional inch movements, when required, should be provided by such items as variable speed or inching motor drives. Control and holding brakes should each be rated at 100% of maximum drive torque at the point application. If two mechanical brakes are provided, one for control and one for holding, they should be adjusted with one brake in each system for both the trolley and bridge leading the other and should be activated by release or shutoff of power. The brakes should also be mechanically tripped to the "on" or "holding" position in the event of a malfunction in the power supply or an overspeed condition. Provisions should be made for manual operation of the brakes. The holding brake should be designed so that is cannot be used as a foot-operated slowdown brake. Drag brakes should not be used. Opposite wheels on bridges or trolleys which support the bridge or trolley on the runways should be matched and have identical diameters. Trolley and bridge speed should be limited. A maximum speed of 30 fpm for the trolley and 40 fpm for the bridge is recommended.

3/ Plugging is the momentary application of full line power to the drive motor for the purpose of promoting a limited movement.

4/ The point in the lowest range of movement control at which power is on, brakes are off, and motors are not energized.

q. The complete operating control system and provisions for emergency controls for the overhead crane handling system should be located in the main cab on the bridge. Additional cabs located on the trolley or lifting devices should have complete control systems similar to the bridge cab. Manual controls for the bridge may be located on the bridge. Remote controls or pendant controls for any of these motions should be the same as those provided in the bridge cab control panel. Provisions should be made in the design for devices for emergency control or operations. Limiting devices, mechanical and electrical, should be provided to indicate, control, and prevent overtravelling and overspeed or hoist (raising or lowering) and for trolley and bridge travel movement. Buffers for bridge and trolley travel should be included.

r. Safety devices such as limit type switches provided for malfunction, inadvertent operation, or failure should be in addition to and separate from the control devices provided for operation.

s. The operating requirements for all travel movements (vertical and horizontal movements, or rotation, singly or in combination) for permanent plant cranes should be clearly defined in the operating manual for hoisting and for trolley and bridge travel. The designer should establish the maximum working load (MWL). The MWL should not be less than 85% of the design rated load (DRL) capacity for the new crane at time of operation. The redundancy provided, design factors, selection of components, and balance of auxiliary-ancilliary and duel items in the design and manufacture should be taken into account in setting the maximum working load for the critical load handling crane system(s). The MWL should not exceed the DRL for overhead crane handling system.

t. When the permanent plant crane is to be used for construction and the operating requirements for construction are not identical to those required for permanent plant service, the construction operating requirements should be defined separately. The crane should be designed structurally and mechanically for the construction loads, plant service loads, and the functional performance requirements for each. At the end of the construction period, the crane handling system should be adjusted for the performance requirements of permanent plant service. The conversion or adjustment may include the replacement of such items as motor drives, blocks, and reeving system. After construction use, the crane should be thoroughly inspected using nondestructive examinations and should be performance tested. If the load and performance requirements are different for construction and plant service periods, then the crane should be tested for both phases. The crane integrity should be verified by the designer and manufacturer and load testing to 125% of the design rated load required for the operating plant should be done before the crane is used as permanent plant equipment.

u. Installation instruction should be provided by the manufacturer. These should include a full explanation of the crane handling system, its controls, and the limitations for the system, and should cover the requirements for installation, testing, and preparation for operation.

4. <u>Mechanical Checks, Testing, and Preventative Maintenance</u>

a. A complete mechanical check of all crane systems as installed should be made to verify the method of installation and to prepare the crane for testing. During and after installation the proper assembly of electrical and structural components should be verified. The integrity of all control, operating, and safety systems is to be verified as to satisfaction of installation and design requirements.

The crane designer and crane manufacturer should provide a manual of information and procedures for use in checking, testing, and crane operation. The manual should also describe a preventive maintenance program based on the approved test results and information obtained during the testing; it should include such items as servicing, repair, and replacement requirements, visual examinations, inspections, checking, measurements, problem diagnosis, nondestructive examination, crane performance testing, and special instructions.

Information concerning proof testing on components and subsystems as required and performed at the manufacturer's plant to verify component or subsystem ability to perform should be available for the checking and testing performed at the place of installation of the crane system.

b. The crane system should be prepared for the static test of 125% of the design rated load. The tests should include all positions of hoisting, lowering, and trolley and bridge travel with the 125% rated load and other positions as recommended by the designer and manufacturer. After satisfactory completion of the 125% static test and adjustments required as a result of the test, the crane handling system should be given full performance tests with 100% of the design rated load for all speeds and motions for which the system is designed. This should include verifying all limiting and safety control devices. The crane handling system should demonstrate the ability to lower and move the design rated load by manual operation and with the use of emergency operating controls and devices which have been included in the handling system.

The complete hoisting machinery should be allowed to two-block during the hoisting test (load block limit and safety devices are bypassed). This test should be conducted without load and at slow speed, to provide assurance of the integrity of the design, equipment, controls, and overload protection devices. The test should demonstrate that when the maximum torque that can be developed by the driving system, including the inertia of the rotating parts at the overtorque condition, will be absorbed or controlled prior to two-blocking.

The complete hoisting machinery should be tested for ability to sustain a load hangup condition by a test in which the load block attaching points are secured to a fixed anchor or excessive load. The drum should be capable of one full revolution before starting the hoisting test.

c. The preventive maintenance program recommended by the designer and manufacturer should also prescribe and establish the MWL for which the crane will be used. The maximum working load should be plainly marked on each side of the crane for each hoisting unit. It is recommended that critical load handling cranes should be continuously maintained at 95% of DRL capacity for the MWL capacity.

C. REFERENCES

1. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel."

2. "Table of Engineering, Manufacturing, and Operating Standards, Practices, and References," attached to this position.

TABLE OF
ENGINEERING, MANUFACTURING, AND OPERATION STANDARDS,
PRACTICES, AND REFERENCES

AISE	Association of Iron and Steel Engineers (Std. No. 6). General items for overhead cranes and specifically for drums, reeving systems, blocks, controls, and electrical, mechanical, and structural components.
AISC	American Institute of Steel Construction, "Manual of Steel Construction." Runway and bridge design loadings for impact, and structural supports.
ASME	American Society of Mechanical Engineers. References for testing, materials, and mechanical components.
ASTM	American Society for Testing Materials. Testing and selection of materials.
ANSI	American National Standards Institute (A10, B3, B6, B15, B29, B30 and N45 series N series of ANSI standards for quality control). ANSI consensus standards for design, manufacturing, and safety.
IEEE	Institute of Electrical and Electronics Engineers. Electrical power and control systems.
AWS	American Welding Society (D1.1.72 - 73/74 revisions). Fabrication requirements and standards for crane structure and weldments.
EEI	Edison Electrical Institute. Electrical systems.
SAE	Society of Automotive Engineers, "Standards and Recommended Practices." Recommendations and practices for wire rope, shafting, lubrication, fasteners, materials selection, and load stability.
CMAA	Crane Manufacturers Association of American (CMAA 70). Guide for preparing functional and performance specifications and component selection.
NEMA	National Electrical Manufacturers Association. Electrical motor, control, and component selections.
WRTB	Wire Rope Technical Board and their manufacturing members. Selection of rope, reeving system, and reeving efficiencies.
MHI	Materials Handling Institute and their member associations and association members such as American Gear Manufacturing Association for gears and gear reducers, Antifriction Bearing Manufacturers Association for bearings selection, etc.
WRC	Welding Research Council, "Control of Steel Construction to avoid Brittle Fracture," and Bulletin #168, "Lamellar Tearing."

January 1975

BRANCH TECHNICAL POSITION APCSB NO. 10-1

DESIGN GUIDELINES FOR AUXILIARY FEEDWATER SYSTEM PUMP DRIVE AND POWER SUPPLY DIVERSITY FOR PRESSURIZED WATER REACTOR PLANTS

A. Background

Heat removal from pressurized water reactor plants following reactor trip and a loss of offsite power is accomplished by the operation of several systems including the secondary system via the steam relief system. In addition, similar capability is required to mitigate the consequences of certain postulated pipe breaks. Such heat removal involves heat transfer from the reactor core to the steam generators resulting in the production of steam which is then released to the atmosphere. In this process it becomes necessary to make up water to the steam generators. This is accomplished by the use of an auxiliary feedwater system which generally consists of redundant components that are powered by both electrical and steam driven sources.

The auxiliary feedwater system functions as an engineered safety system because it is the only source of makeup water to the steam generators for decay heat removal when the main feedwater system becomes inoperable. It must, therefore, be designed to operate when needed, using the principles of redundancy and diversity in order to accomplish its function under postulated accident conditions. Current systems are either powered by electrical or steam driven sources. Operating experience demonstrates that each type of motive power can be subject to either a failure of the component itself, its source of energy or its

associated control system. This type of failure can be minimized by the utilization of diverse systems including the energy sources of at least two different and distinct varieties.

The choice of several independent flow paths for the auxiliary feedwater system serves to preclude the possibility of a complete loss of function due to a single event, either occurring alone, or in conjunction with the failure of an active component. The auxiliary feedwater system is categorized as a high energy system, because either that section of line which joins the main feedwater system or the steam generator is pressurized during plant operation or else the entire system is pressurized when in use during startup, hot standby, and shutdown.

The Auxiliary and Power Conversion System Branch believes that it is necessary to set forth acceptable design guidelines for the auxiliary feedwater system, and in this regard has developed acceptable positions that may be used to select the minimum diversification of auxiliary feedwater system components required for the long-term cooling of the reactor facility.

B. Branch Technical Position

1. The auxiliary feedwater system should consist of at least two full capacity independent systems including diverse power sources.
2. Other powered components of the auxiliary feedwater system should likewise share in the concept of separate and multiple sources of motive energy. (An example of the required diversity would be two separate auxiliary feedwater systems, each capable of removing the decay heat load of the reactor system, having

one separate system powered from either of two AC sources and the other system wholly powered by steam and DC electric power.)

3. The piping arrangement, both intake and discharge, for each system should be designed to permit the pumps to supply feedwater to any combination of steam generators. This should take into account pipe failure, active component failure, power supply failure, control system failure that would defeat the diversity requirement. One method that would be acceptable is crossover piping containing valves that can be operated remote manually from the control room using the power diversity principle for the valve operators and actuation systems.

4. The auxiliary feedwater system should be designed with suitable redundancy to offset the consequences of any single active component failure; however, each subsystem need not contain redundant active components.

5. When considering a high energy line break, the system should be so arranged as to permit the capability of supplying necessary emergency feedwater to the steam generators, despite the postulated rupture of any high energy section of the system, assuming a concurrent single active failure.

DEC 2 8 1976

Docket Nos. 50-361/362

MEMORANDUM FOR: K. Kniel, Chief, Light Water Reactors Branch 2, DPM

FROM: W. E. Kreger, Chief, Radiological Assessment Branch, DSE

SUBJECT: SAN ONOFRE 2&3, ACCEPTANCE REVIEW

PLANT NAME: San Onofre 2&3
LICENSING STAGE: FSAR Acceptance Review
DOCKET NUMBERS: 50-361/362
MILESTONE NUMBER/BRANCH CODE: 01-33
PROJECT MANAGER: H. Rood
RESPONSIBLE BRANCH: LWR-2
REQUESTED COMPLETION DATE: 12/23/76
DESCRIPTION OF RESPONSE: Acceptance Review
REVIEW STATUS: Continuing

The Radiation Protection Section of the Radiological Assessment Branch has completed its review of the San Onofre 2&3 FSAR and recommends that the FSAR be accepted.

Requests for additional information needed for program review are enclosed.

This review was performed by C. S. Hinson, RPS/RAB.

Original signed by
W. E. Kreger
William E. Kreger, Chief
Radiological Assessment Branch
Division of Site Safety and
Environmental Analysis

Enclosure: as stated

cc w/o encl.:
W. McDonald
P. Shuttleworth

cc w/encl.: (cont'd on attached)
S. Hanauer
H. Denton
D. Muller
F. Miraglia
R. Vollmer
J. Miller
S. Varga
R. Boyd

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50-362
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DATE ➤	X12/21/76	12/21/76	12/22/76			

K. Kniel

DEC 22 1976

cc w/encl.:
J. Collins
H. Rood
T. Murphy
N. Dayem
R. Emch
J. Nehemias
C. Hinson

OFFICE ➤						
SURNAME ➤						
DATE ➤						

331.0 RADIATION PROTECTION

331.1 (12.2.2.1) The term ($V\lambda_{Ti}$) should appear in the numerator, not the denominator, of the first equation in section 12.2.2.1.

331.2 (Table 12.2-6) Give the units of the figures in Table 12.2-6.

331.3 (Table 12.2-8) Include expected particulate isotopes in your table of normal airborne radioactivity concentrations (Table 12.2-8).

331.4 (12.3.1) Adequate and rapidly serviceable lighting should be provided for each room or cubicle containing zone III-V components. State how you plan to implement the above.

331.5 (Table 12.4.9) Do the figures in the last four columns of Table 12.4-9 represent the doses received by individual plant employees working in the areas listed, or do they represent the total accumulated personnel dose commitment caused by airborne radioactivity in these areas? If these figures represent individual worker doses, list the total number of personnel expected to work in each of the areas listed and give the total estimated personnel dose caused by airborne radioactivity.

331.6 (12.5.2.1) On figures 12.3-1 through 12.3-25 indicate the major traffic patterns used by plant personnel during their daily activities. Also describe the route a plant employee would take in going from the main entrance to the controlled area.

331.7(a) (12.5.2.2.5) The airborne radioactivity monitoring system should be sensitive enough to indicate that an airborne radioactivity hazard exists in any compartment (or area) for which the monitor is applicable. Assume a 1 MPC concentration of the most representative particulate and gas is present in the compartment with the lowest flowrate in the area being monitored during normal operation. For each airborne radioactivity monitor in the plant, give the response time to detect this concentration.

331.7(b) (12.5.2.2.5) In order to adequately detect airborne radioactivity in areas which may be occupied by personnel, airborne radioactivity monitors should be located upstream of the air cleaning systems. It is not clear from studying Fig. 9.4-9 (sheet 2) whether this is being done for the exhaust air from the fuel handling building. Provide assurance that all airborne radioactivity monitors sampling air from areas which may be occupied by personnel have sampling points upstream of the air cleaning systems. Provide HVAC drawing 40090.

Docket
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50-362

DEC 22 1976

Docket Nos. 50-361/362

MEMORANDUM FOR: K. Kniel, Chief, Light Water Reactors Branch No. 2, DPM

FROM: J. Collins, Chief, Effluent Treatment Systems Branch, DSE

SUBJECT: ACCEPTANCE REVIEW OF THE FINAL SAFETY ANALYSIS REPORT FOR SAN ONOFRE, UNIT NOS. 2 AND 3

PLANT NAME: San Onofre Nuclear Generating Station
LICENSING STAGE: OL
DOCKET NUMBERS: 50-361/362
MILESTONE NUMBER: None
RESPONSIBLE BRANCH: LWR No. 2
PROJECT MANAGER: H. Rood
DESCRIPTION OF RESPONSE: Final Safety Analysis Report Acceptance Review
REQUESTED COMPLETION DATE: December 22, 1976
REVIEW STATUS: Additional Information Needed

We have completed our acceptance review of Sections 6.5.1, 9.2.4, 9.3.2, 10.4.2, 10.4.3, 10.4.8, 11.1.5, and 15.7.3 of the Final Safety Analysis Report for the San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, and find they are approximately 95% complete. However, we will need additional information concerning hydrogen and oxygen analyzers and their functions to prevent hydrogen explosions in the gaseous radwaste system and provisions for monitoring potentially radioactive liquid waste release points. A list of the required information is enclosed. We recommend that the application be accepted for review.

The meteorological information noted in the applicant's letter, dated November 30, 1976, however, is scheduled to be submitted in April 1977. Since this information is needed in order to complete the Appendix I evaluation of the plant radwaste systems, the safety review schedule should consider the April 1977 date in establishing the schedule for ETSB input to the SER.

ORIGINAL SIGNED BY
JOHN T. COLLINS

John T. Collins, Chief
Effluent Treatment Systems Branch
Division of Site Safety and
Environmental Analysis

Enclosure:
Additional Questions

50-362
M 4

OFFICE ➤						
SURNAME ➤						
DATE ➤						

cc: S. Hanauer
H. Denton
D. Muller
J. Miller
R. Boyd
R. Vollmer
V. Moore
W. Regan
O. Lynch
D. Bunch
W. Kreger
D. Vassallo
H. Rood
W. McDonald (w/o encl)
J. Panzarella (w/o encl)
P. Shuttleworth (w/o encl)
W. Burke
P. Stoddart

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JTCollins

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x 27775 SURNAME ➤	PStoddart:do	WBurke	JCollins			
DATE ➤	12/21/76	12/21/76	12/21/76			

1. (9.2.1.5, 11.5) You state that the discharge from the saltwater cooling system is not monitored for radioactivity content. You should provide continuous monitoring of gross radioactivity concentration for this potentially radioactive discharge release.

The occurrence of radioactivity concentrations exceeding a predetermined level should be alarmed at the reactor control room.

2. (11.3.1.6, 9.3.2) In Subsection 11.3.1.6, "Hydrogen Control", you state that hydrogen and oxygen analyzers in the gaseous radwaste system initiate alarms at predetermined setpoints prior to reaching a potentially explosive hydrogen-oxygen mixture and that manual action by the operator is required (to correct abnormal conditions). For systems not designed to withstand a hydrogen explosion, you should provide hydrogen or oxygen analyzers which actuate automatic control functions to preclude the formation or buildup of explosive hydrogen-oxygen mixtures. You also state that the waste gas surge tank is sampled and analyzed intermittently on a timed cycle and that other points are selected manually.

For systems not designed to withstand a hydrogen explosion, you should provide, in addition to the timed-cycle analyzers described in Subsection 9.3.2, a continuously operating hydrogen or oxygen analyzer on the line between the compressor outlet and the waste gas decay tank inlet; this analyzer should also have the automatic control functions described above.

3. (11.5) In Table 11.5-1, "Continuous Process and Environmental Radiation Monitoring," the concentration values for range, expected concentrations, and alarm setpoint are shown in "Ci/cm^3"; these appear to be typographical errors and should be changed to read either "uCi/cm^3" or "Ci/m^3". Also in Table 11.5-1, the range of the Condenser Air Ejector High Range Monitor is given as 10^{-2}(u)Ci/cm^3; this also appears to be a typographical error and should read either $10^2 uCi/cm^3$ or $10^2 Ci/m^3$.

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J. Lee

DOCKET NOS. 50-361
AND 50-362

NOTE FOR: Harry Rood, Light Water Reactors Branch No. 2, DPM

FROM: Jean Lee, Licensing Assistant, Light Water Reactors Branch No. 2, DPM

SUBJECT: REVIEW OF GENERAL INFORMATION - SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 and 3

I have reviewed the General Information of the application tendered by Southern California Edison Company and San Diego Gas and Electric Company for operating licenses for the San Onofre Nuclear Generating Station, Units 2 and 3. Although the General Information is acceptable for docketing, there are certain areas where more information is needed.

1. The discussion on page 6 (item 7) concerning financial qualifications contains a minimum of information. I have discussed this with one of our financial analysts and he indicated that additional information will probably be required at a later time in order to make a determination on financial qualifications of the applicants. This additional information will be based on the requirements of 10 CFR 50.33(f) and Appendix C to 10 CFR 50.

2. Prior to docketing, the Application must be signed by both applicants and notarized.

3. The General Information should state the full name, title and address we should use when writing to the applicants. Also, if copies of our correspondence is to be forwarded to a key person within the organizations of the applicants and/or the applicants' counsel, it should be indicated in the General Information.

50-362
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SURNAME ➤						
DATE ➤						

Our regulations contained in 10 CFR 50.30(c)(1)(i) require applicants to retain copies of their applications for distribution in accordance with our instructions. Enclosed is the distribution list for service of the General Information, Safety Analysis Report and amendments.

/s/

Jean Lee,
Licensing Assistant
Light Water Reactors Branch No. 2
Division of Project Management

cc: KKniel
MMlynczak

OFFICE➤	DPM:LWR #2					
SURNAME➤	JLee:ab					
DATE➤	12/20/76					

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APPLICATION, SAFETY ANALYSIS REPORT AND AMENDMENTS

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P. O. Box X
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket Nos. 50-361/362

DEC 15 1976

MEMORANDUM FOR: D. B. Vassallo, Assistant Director for LWRs, DPM

THRU: T. M. Novak, Chief, Reactor Systems Branch, DSS

FROM: S. L. Israel, Section Leader, Reactor Systems Branch, DSS

SUBJECT: SAN ONOFRE UNIT NOS. 2 AND 3 ACCEPTANCE REVIEW

Plant Name:	San Onofre, Units 2 and 3
Docket Numbers:	50-361 and 50-362
Licensing Stage:	OL
Milestone Number:	01-21
Responsible Branch and Project Manager:	LWR-2 Harry Rood
Systems Safety Branch Involved:	Reactor Systems Branch
Description of Review:	Acceptance Review
Requested Completion Date:	December 7, 1976
Review Status:	Complete

The Reactor Systems Branch has reviewed Sections 3.5.1.2, 4.6, 5.2.2, 5.2.7, 5.5.7, 6.3, 6.x, and 15 of the San Onofre draft FSAR. These sections are acceptable for docketing.

Sanford Israel

Sanford L. Israel
Section Leader
Reactor Systems Branch
Division of Systems Safety

cc: S. Hanauer
R. Heineman
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49-27591

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

DEC 15 1976

Docket Nos. 50-361/362

MEMORANDUM FOR: Domenic B. Vassallo, A/D for LWRs, DPM

FROM: Laurence E. Phillips, Section Leader, Reactor Analysis Section, Analysis Branch, DSS

THRU: Zoltan R. Rosztoczy, Chief, Analysis Branch, DSS ZRR

SUBJECT: ACCEPTANCE REVIEW OF SAN ONOFRE 2 AND 3, DOCKET NOS. 50-361/362

The Reactor Analysis Section of the Analysis Branch has reviewed Section 4.4 of the San Onofre 2/3 FSAR in connection with the requested acceptance review. We find that the information supplied in Section 4.4 as detailed in the FSAR is adequate for acceptance for docketing.

G. N. Lauben for
Laurence E. Phillips, Section Leader
Reactor Analysis Section
Analysis Branch
Division of Systems Safety

cc: Z. Rosztoczy
L. Phillips
H. Rood
M. McCoy
M. Mlynczak

Contact: M. McCoy, NRR, 27911

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Distribution
Docket File
LWR #2 File
HRood
MMlynczak
KKniel

Docket Nos. 50-361
and 50-362

MEMORANDUM FOR: J. Knight, Assistant Director for Engineering, DSS
D. Ross, Assistant Director for Reactor Safety, DSS
R. Tedesco, Assistant Director for Plant Systems, DSS
R. Vollmer, Assistant Director for Site Analysis, DSE
W. Gammill, Assistant Director for Site Technology, DSE
D. Skovholt, Assistant Director for Quality Assurance and Operations, DPM
J. Saltzman, Chief, Antitrust and Indemnity Group, NRR
G. Roy, Chief, Field Coordination and Enforcement Branch, I&E

FROM: Harry Rood, Project Manager, Light Water Reactors Branch No. 2, DPM

SUBJECT: ACCEPTANCE REVIEW OF FSAR FOR SAN ONOFRE NUCLEAR GENERATING STATION (SONGS), UNITS 2 & 3

Southern California Edison Company and San Diego Gas & Electric Company have tendered their application on December 1, 1976 for Operating Licenses for two pressurized water nuclear reactors at their site in San Diego County, California, adjacent to San Onofre Unit 1. Each NSSS will have a rated output of 1140 electrical megawatts.

Participants in the acceptance review should submit their evaluation to me by December 22, 1976. Reviewers are requested to advise me when they start their review and to notify me of significant omissions or unusual problems prior to completion of the evaluation memoranda. We expect to notify the applicant of the results of the acceptance review on January 7, 1977 and will schedule a meeting with them shortly thereafter. You will be requested to discuss your acceptance review evaluation with the applicant at that meeting.

Ms. Margaret Mlynczak has recently joined our Branch as a Licensing Project Manager and will be participating in the SONGS 2/3 acceptance review. Please contact her if I am not available.

Original Signed by
Harry Rood

H. Rood, Project Manager
Light Water Reactors
Branch No. 2
~~Division of Project Management~~

50-362
M4

cc: See page 2

OFFICE ➤		DPM:LWR #2				
SURNAME ➤		HRood:mt				
DATE ➤		12/ /76				

cc: AD, Safety
AD, Environmental
DSS BC's
QA&O BC's
D. Bunch
J. Collins
W. Kreger
J. Stepp
L. G. Hulman
E. Markee
L. Heller
J. Goll
R. Ballard
B. J. Youngblood
EPM
LA's (safety and environmental)
J. Panzarella
P. Shuttleworth
S. A. Treby
L. Chandler

SEP 1 6 1976

DOCKET NOS. 50-361
and 50-362

APPLICANT: SOUTHERN CALIFORNIA EDISON COMPANY (SCE)

FACILITY: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

SUMMARY OF MEETING HELD ON SEPTEMBER 8, 1976 CONCERNING SAN ONOFRE METEOROLOGY

On September 8, 1976, the NRC staff met in Bethesda, Maryland with representatives of SCE, and their consultants, to discuss the above subject. A list of the attendees at the meeting is included as Enclosure 1.

At the meeting, the applicant summarized its proposed onshore tracer test program to be conducted at the San Onofre site. The material presented by the applicant is included as Enclosure 2.

Following the presentation, the staff suggested that the test program should remain flexible, in terms of the number of tests conducted under each set of test conditions, until the initial results are obtained and evaluated. Then, the remainder of the tests could be oriented towards those test conditions which most effectively accomplish the test objectives; in particular, demonstrating the adequacy of using bluff meteorology to estimate dispersion. The staff also stated that in order to show that a permanent onshore meteorological tower is not necessary, the tracer tests must establish that a consistent correlation exists between the X/Q values obtained from the tracer tests and those predicted from bluff tower data.

It was agreed at the meeting that any additional staff comments would be given to the applicant by September 15, 1976. The applicant's target date for submittal of a report describing the proposed tests is September 20, 1976.

S/

Harry Rood
Light Water Reactors
Branch No. 2
Division of Project Management

Enclosures & ccs:
See page 2

OFFICE ➤	DPM:LWR #2					
SURNAME ➤	HRood:mt					
DATE ➤	9/ /76					

Enclosures:
1. Attendance list
2. Material presented by the applicant

cc: Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas and Electric Company
ATTN: Mr. Jack E. Thomas
Vice President - Electric
101 Ash Street
P. O. Box 1831
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Rollin E. Woodbury, General Counsel
Southern California Edison Company
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Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas and Electric Company
111 Sutter Street
San Francisco, California 94104

Mr. David Sakai
845 North Perry Avenue
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Frederick P. Sutherland, Esq.
Center for Law in the Public Interest
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Mr. Kenneth E. Carr
City Manager
City of San Clemente
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Alan R. Watts, Esq.
Assistant City Attorney
City Hall
Anaheim, California 92805

Lawrence Q. Garcia, Esq.
California Public Utilities Commission
5066 State Building
San Francisco, California 94102

George Spiegel, Esq.
2600 Virginia Avenue, N. W.
Washington, D. C. 20036

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ENCLOSURE 1

ATTENDANCE LIST
MEETING WITH SOUTHERN CALIFORNIA EDISON COMPANY
SAN ONOFRE NUCLEAR GENERATING STATION
SEPTEMBER 8, 1976

NUS
E. Mitchell
M. Septoff
J. H. Taylor

SCE
W. C. Moody
D. E. Nunn
L. J. Brunton

SDG&E
W. D. Griffith

SAI
Lynn Tevschev

NRC - STAFF
H. Rood
E. H. Markee
A. Burger
R. W. Froelich
A. Schwencer
L. G. Hulman

OFFICE ➤						
SURNAME ➤						
DATE ➤						

AGENDA

Onshore Tracer Test Program
San Onofre Nuclear Generating Station
September 8, 1976

I. INTRODUCTION

II. DESCRIPTION OF PROPOSED TRACER TEST PROGRAM

A. Objectives

B. Site Description

C. Test Design - Tracer Release and Collection

D. Test Design - Meteorology

E. Data Analyses

III. ANNUAL REPRESENTATIVENESS OF TRACER TESTS

IV. LOCATING THE INLAND TOWER PERTAINING TO THE LAND-SURFACE INTERNAL BOUNDARY LAYER

V. CONCLUSIONS

ONSHORE TRACER PROGRAM

- OBJECTIVES
- SITE DESCRIPTION
- TEST DESIGN - TRACER RELEASE AND COLLECTION
- TEST DESIGN - METEOROLOGY
- DATA ANALYSES

OBJECTIVES

- MEASURE AND CHARACTERIZE ATMOSPHERIC DISPERSION TO PERMIT REALISTIC CALCULATIONS OF SHORT TERM ACCIDENT DISPERSION FACTORS
- DEMONSTRATE APPROPRIATENESS OF USING BLUFF METEOROLOGY TO ESTIMATE DISPERSION

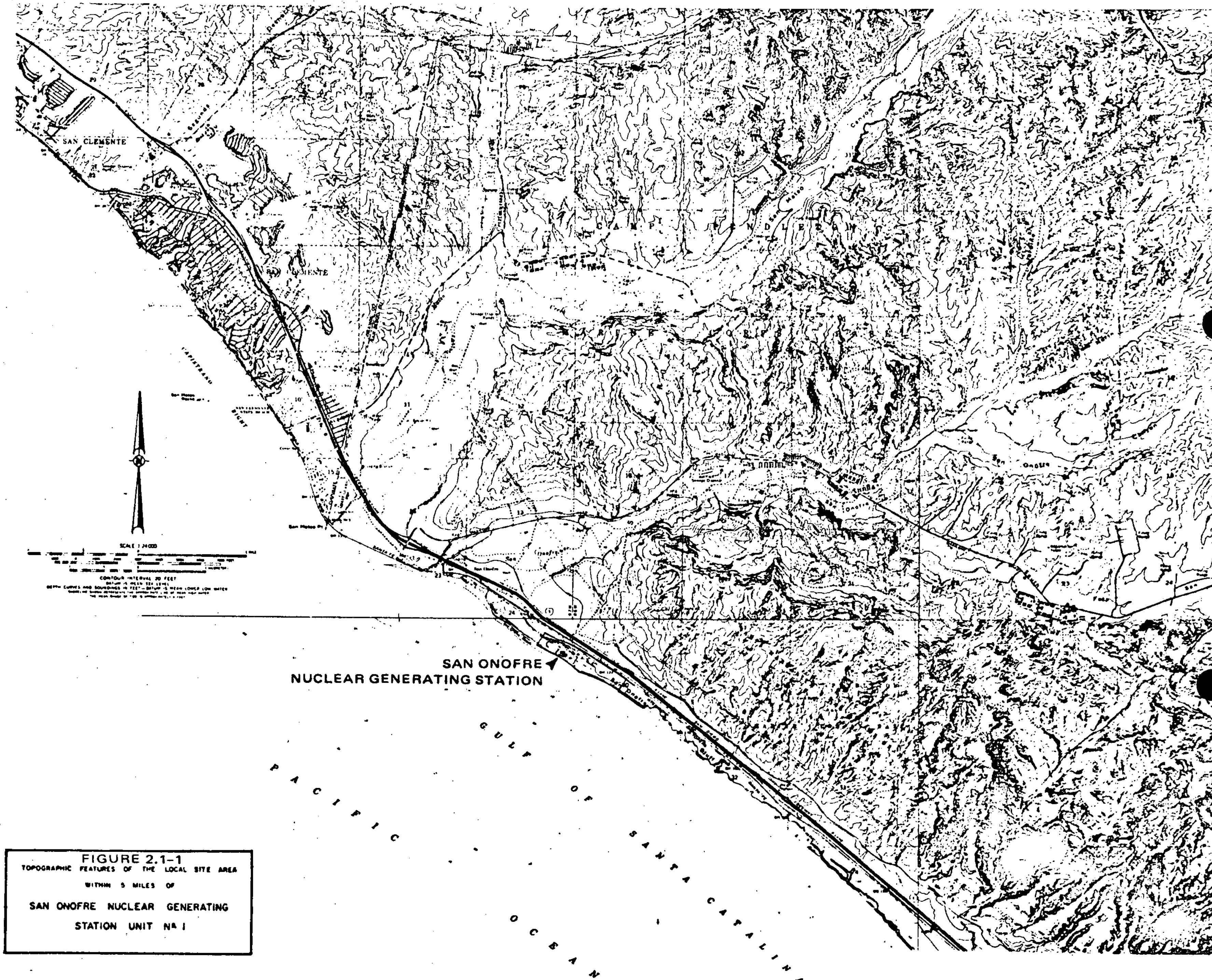

FIGURE 2.1-1
TOPOGRAPHIC FEATURES OF THE LOCAL SITE AREA
WITHIN 5 MILES OF
SAN ONOFRE NUCLEAR GENERATING
STATION UNIT No I

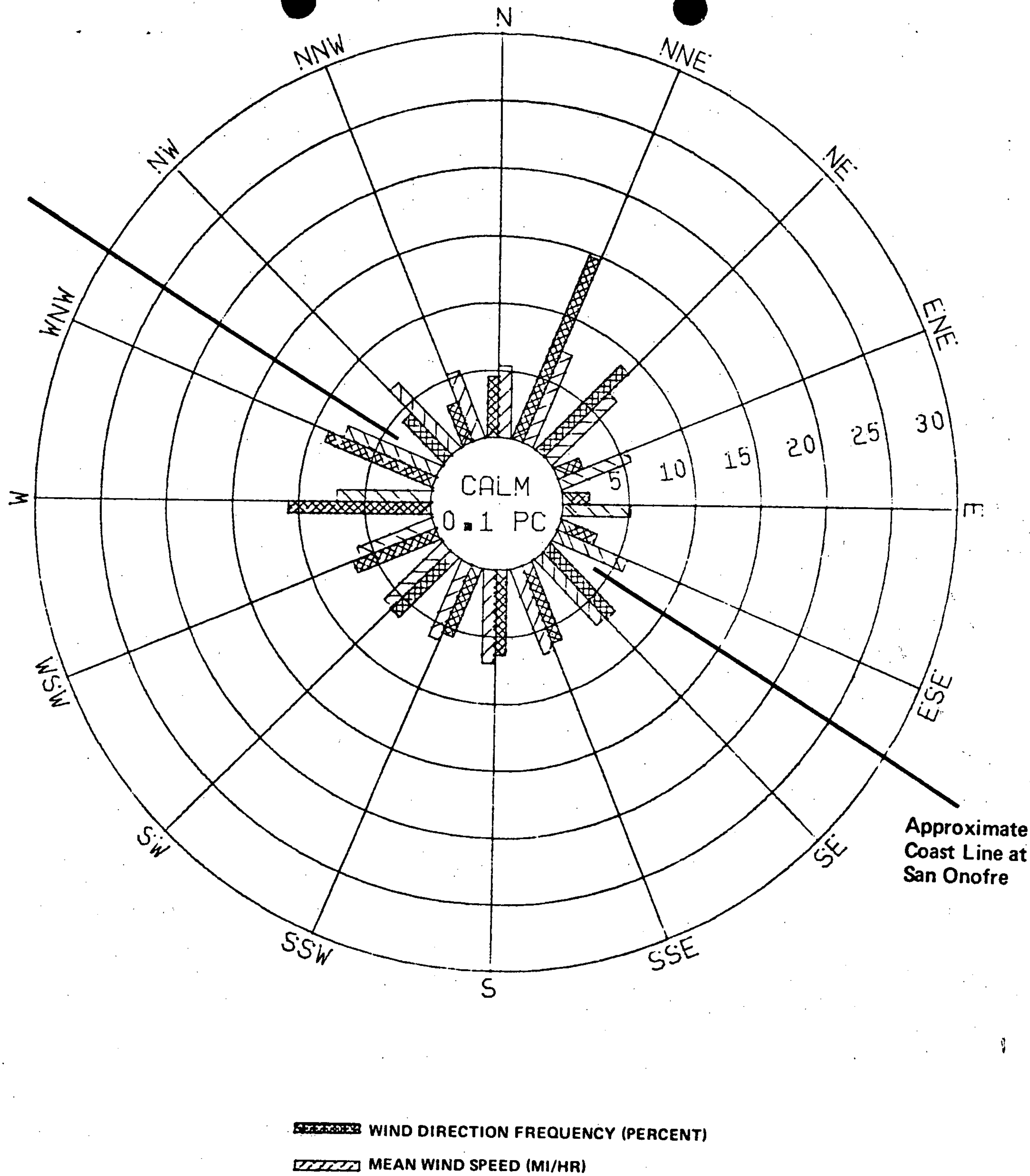

FIGURE 2.2–1
ANNUAL WIND ROSE FOR SONGS BASED ON
THE IO METER LEVEL BLUFF TOWER (1/25/75 – 1/24/76)

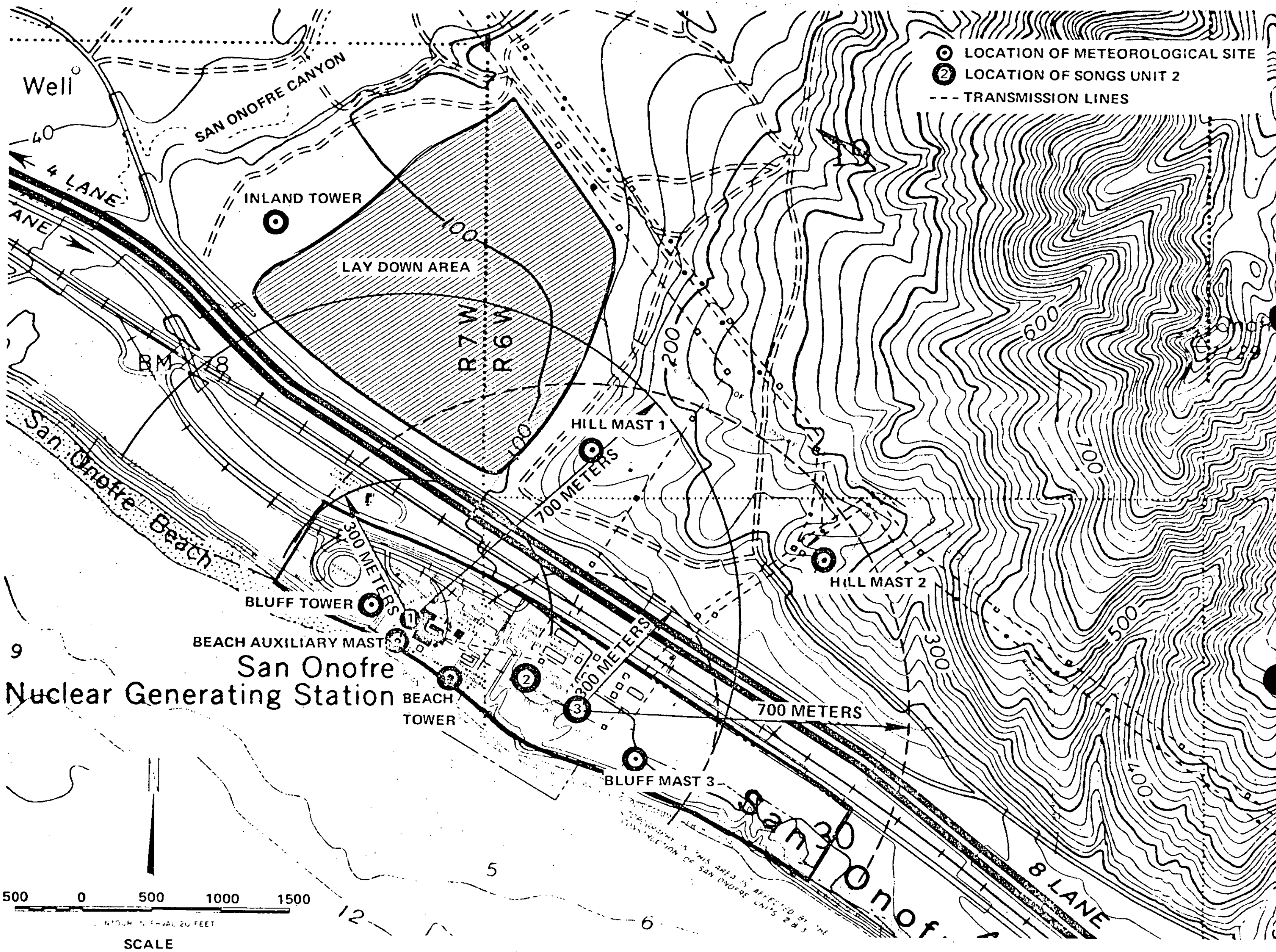

FIGURE 4.1-1 SONGS METEOROLOGICAL INSTRUMENTATION LOCATIONS

TRACER RELEASE

- ANTICIPATED USE OF DUAL TRACER TESTS
- ELEVATIONS APPROXIMATELY AT GROUND LEVEL BELOW BLUFF AND TOP OF DOME
- SMOKE AT RELEASE POINT
- UNIT 1 AND UNITS 2 OR 3

SAMPLE COLLECTION

- AUTOMATIC AND SEMI-AUTOMATIC BAG SAMPLES
- TWO SETS OF SAMPLING ARCS
- 700 METERS AND 300 METERS

SAMPLE ANALYSIS

- SIMILAR TO BEACH TRACER TESTS

TEST DESIGN METEOROLOGY

- o TARGET MIX OF METEOROLOGICAL CONDITIONS
  - oo ONSHORE WIND FLOW
  - oo LOW TO MODERATE WIND SPEEDS
  - oo NEUTRAL TO STABLE STABILITY

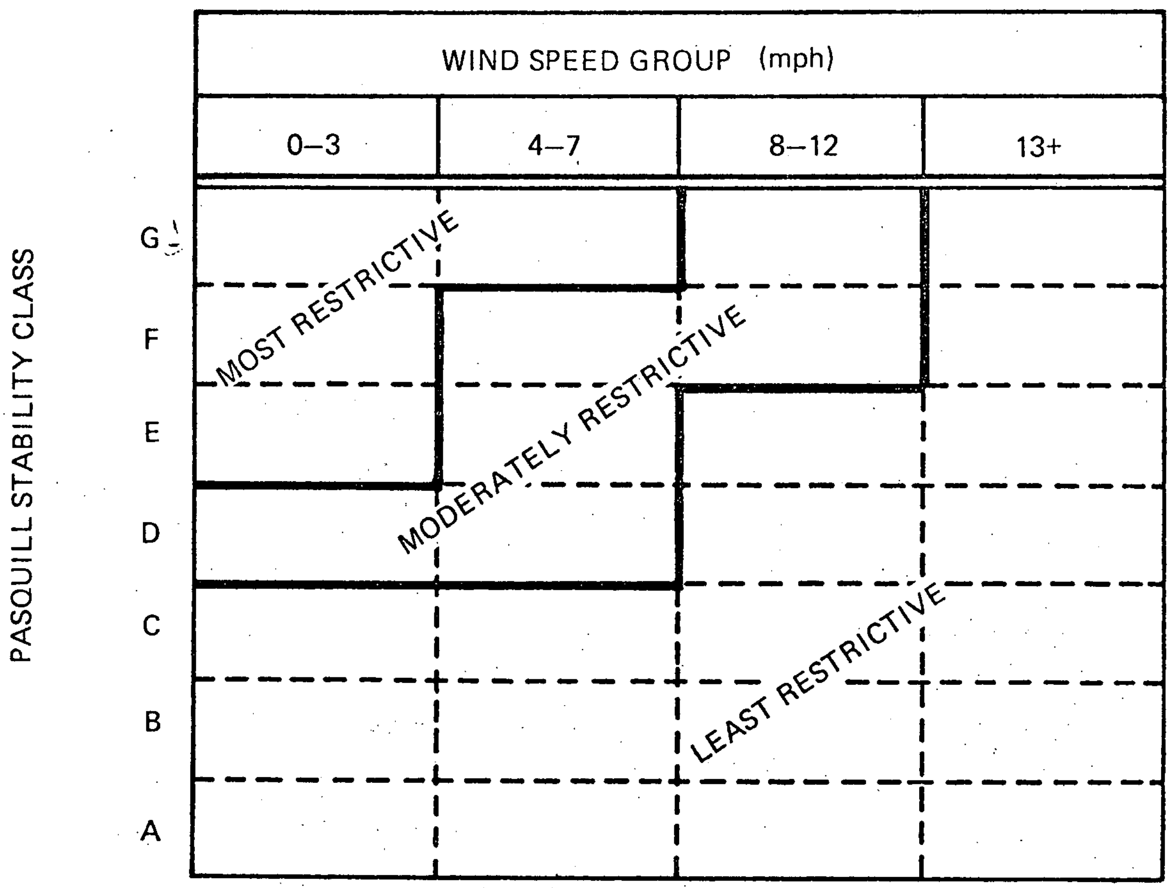

TABLE 3.1-1 DILUTION POTENTIAL CLASSES

TABLE 3.1-2

PERCENT FREQUENCY OF OCCURRENCE IN WINTER
OF WIND DIRECTION VS. DILUTION POTENTIAL CLASS AT SONGS

The values are based on meteorological data from the main bluff tower, 10 meter winds and ΔT(120ft-20ft) for the period of February 1, 1975 to January 28, 1976 (SCE, 1976). Winter includes the months of December, January, and February. Values may not total exactly due to rounding.

Dilution Potential Class	Alongshore Flow-Southerly (SE-SSE)	Direct Onshore Flow (S-WSW)	Alongshore Flow-Northerly (W-WNW)
Most Restrictive	4	3	2
Moderately Restrictive	4	3	3
Least Restrictive	6	8	11

DESIGN NUMBER OF TESTS PER UNIT

Dilution Potential Class	Alongshore Flow-Southerly (SE-SSE)	Direct Onshore Flow (S,SSW,SW,WSW)	Alongshore Flow-Northerly (W-WNW)
Most Restrictive	6	6	6
Moderately Restrictive	6	6	6
Least Restrictive	6	6	6

DATA ANALYSES

DESIGNED TOWARDS QUANTIFYING
THE OBJECTIVES OF THE PROGRAM

TYPES OF ANALYSES

- EVALUATION OF GAUSSIAN NATURE OF PLUMES
- SUMMARIZATION OF METEOROLOGY AND TRACER CONCENTRATIONS
- COMPARISON OF MEASURED AND CALCULATED CONCENTRATIONS
- INVESTIGATING USE OF ALTERNATE FORMULATIONS
- DETERMINING SITE SPECIFIC DISPERSION PARAMETERS

REASONS FOR CONCLUSION REGARDING SEASONAL REPRESENTATIVENESS

- o FREQUENCY OF DILUTION CLASSES
- o FACTORS DOMINATING YEAR ROUND WIND FLOW
- o LAND/SEA TEMPERATURE DIFFERENCES
- o OFFSHORE AIR/SEA TEMPERATURE DIFFERENCE
- o OVERWATER FETCH TEMPERATURES
- o COASTLINE VARIABILITY OF SEA SURFACE TEMPERATURE
- o OBSERVED VERTICAL TEMPERATURE PROFILES

TABLE 3.2-1

FREQUENCY (%) OF OCCURRENCE OF DILUTION CLASSES AT SONGS FOR ONSHORE FLOW

The Values are based on the Main Bluff Tower data, 10 m winds and $\Delta T_{120ft-20ft}$, from SCE, 1976 for the period of February 1, 1975 to January 28, 1976. Winter includes the months of December, January, and February. The frequency, rounded to the nearest whole percent, is based on the total number of observations for all directions during the time period. Values may not necessarily total exactly due to rounding. Onshore flow includes the directions of southeast thru west-northwest.

Dilution Class	Time Period: Annual	Time Period: Winter
Most Restrictive	4%	8%
Moderately Restrictive	10%	10%
Least Restrictive	44%	26%
All Classes	58%	45%

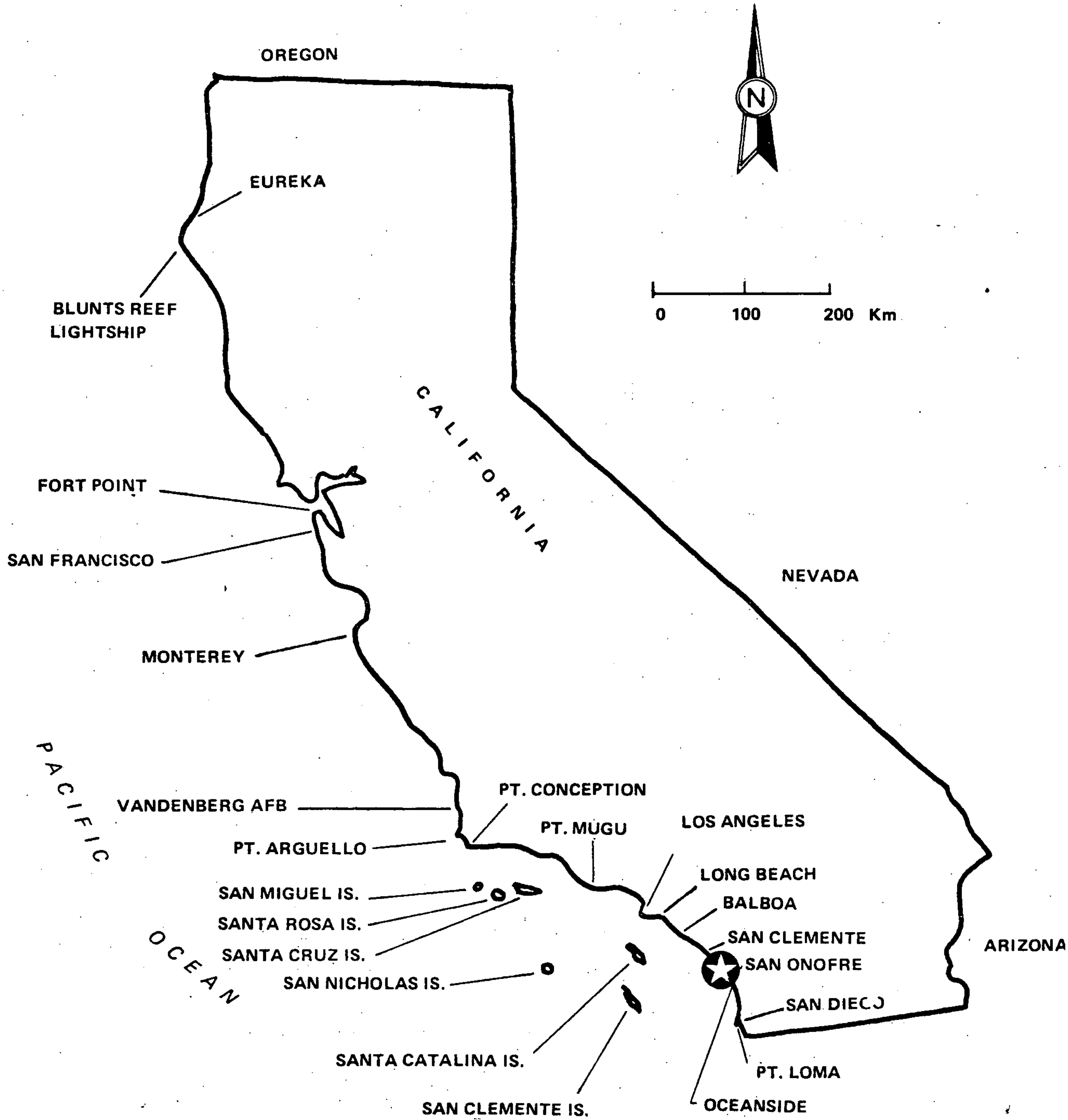

FIGURE 3.2-1

MAP OF CALIFORNIA AND ADJACENT COASTAL WATERS

TABLE 3.2-3

OFFSHORE AND UPWIND SEA SURFACE TEMPERATURES FOR SAN ONOFRE

The "offshore" gridpoint of 118°W longitude and 33.5°N latitude, about 20 km offshore of SONGS was selected; the ocean is about 180 m deep at this point. The "upwind" gridpoint of 121.5°W longitude and 34°N latitude about 230 km west-northwest of the "offshore" point was selected; here, south of Pt. Conception just west of San Miguel Island the water is also 180 m deep or more. The average temperature data are based on ship reports (Naval Weather Service Command, 1971). Data were read to the nearest ½ degree Fahrenheit. Values may not necessarily total exactly due to rounding.

Time Period	Offshore Temp. (°F)	Upwind Temp. (°F)	Offshore Minus Upwind Temp. (°F)
January	59.0	57.0	2.0
February	59.0	56.5	2.5
March	59.0	56.0	3.0
April	60.0	55.5	4.5
May	61.0	56.5	4.5
June	63.5	58.5	5.0
July	65.0	60.5	4.5
August	69.0	62.0	7.0
September	68.0	63.5	4.5
October	66.5	62.0	4.5
November	63.5	60.5	3.0
December	60.5	58.5	2.0
Annual	62.8	58.9	3.9 (~2°C)

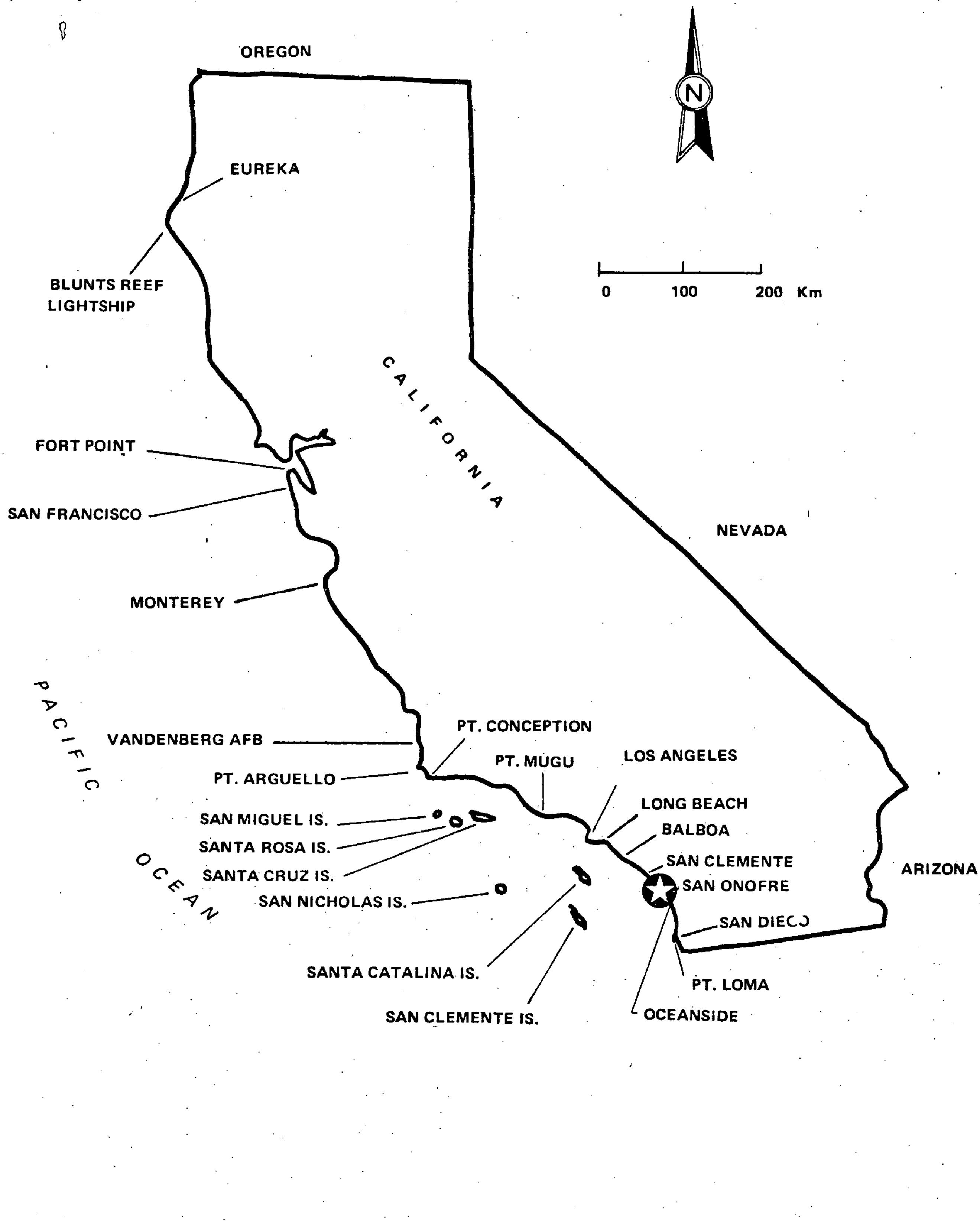

FIGURE 3.2-1

MAP OF CALIFORNIA AND ADJACENT COASTAL WATERS

SEASONAL REPRESENTATIVENESS

THE RESULTS OF AN ONSHORE TRACER PROGRAM UNDER VARIOUS METEOROLOGICAL CONDITIONS AT SAN ONOFRE IN WINTER ARE JUDGED TO BE GENERALLY APPROPRIATE TO REPRESENT DISPERSION CONDITIONS THROUGHOUT THE YEAR

TABLE 3.2-2

SOUTHERN CALIFORNIA OFFSHORE AIR AND SEA SURFACE TEMPERATURES

The "offshore" gridpoint of 118°W longitude and 33.5°N latitude about 20 km offshore of SONGS was selected; the ocean depth is about 180 m at this point. The average temperature data are based on ship reports (Naval Weather Service Command, 1971). Data were read to the nearest ½ degree Fahrenheit. Values may not necessarily total exactly due to rounding.

Time Period	Air Temp. (°F)	Sea Surface Temp. (°F)	Air Minus Sea Temp. (°F)
January	59.0	59.0	0
February	57.5	59.0	-1.5
March	59.0	59.0	0
April	59.5	60.0	-0.5
May	61.0	61.0	0
June	63.0	63.5	-0.5
July	65.5	65.0	+0.5
August	68.5	69.0	-0.5
September	68.0	68.0	0
October	66.0	66.5	-0.5
November	62.0	63.5	-1.5
December	59.5	60.5	-1.0
Annual	62.4	62.8	-0.5 (~-0.25°C)

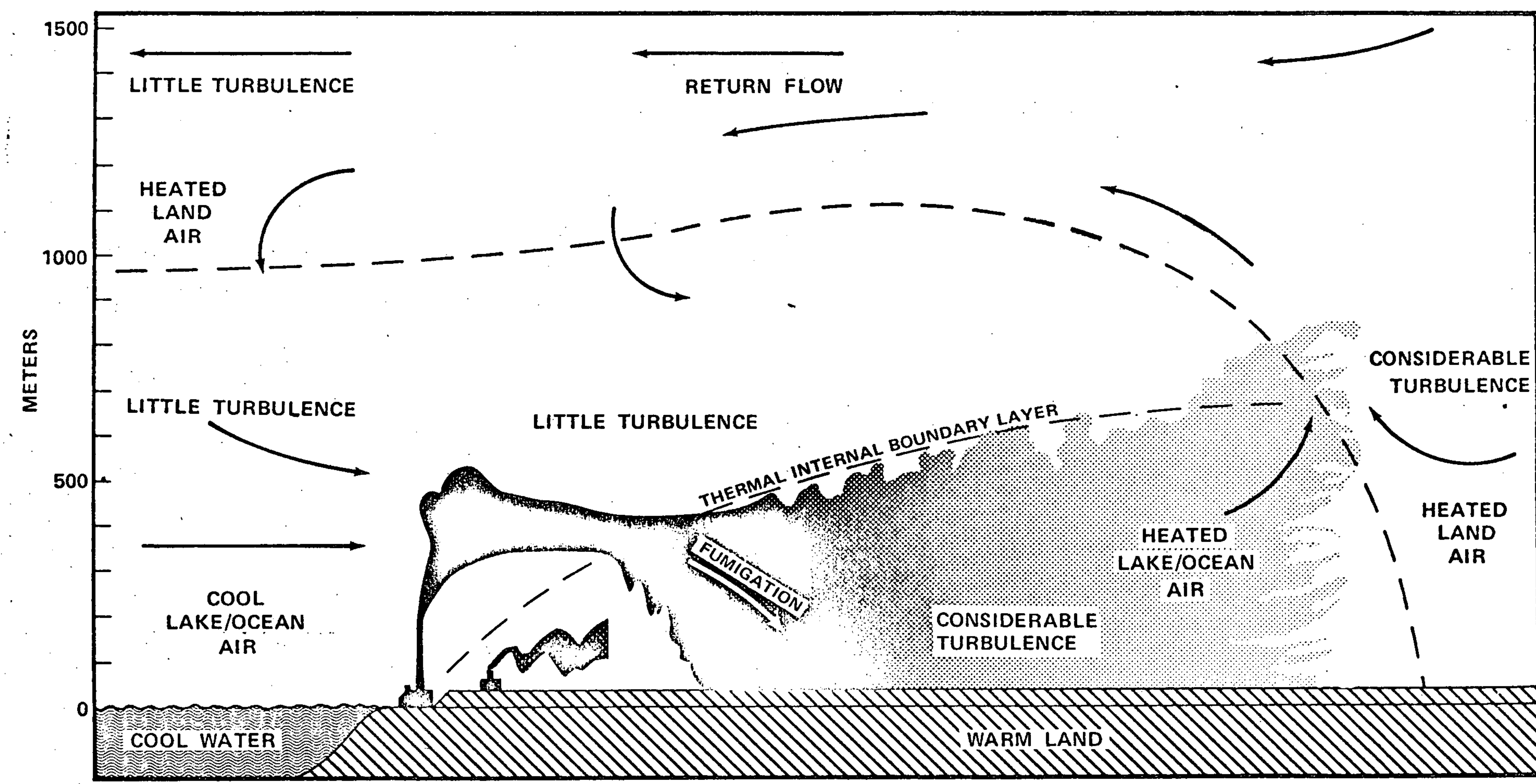

Figure Lake/Sea Breeze Circulation And Its Effect On Near-Coastal Releases

TABLE A.2-1

THERMAL INTERNAL BOUNDARY LAYER DEPTH AFTER RAYNOR, EQN (A.2-1)

a. $\partial T/\partial Z = +0.001^{\circ}C/m$

Inland Fetch (m)	$\lvert\theta_1-\theta_2\rvert = 2^{\circ}C$	$\lvert\theta_1-\theta_2\rvert = 6^{\circ}C$
500	100m	173m
750	122	212
1000	141	245
1500	173	300

b. $\partial T/\partial Z = +0.005^{\circ}C/m$

Inland Fetch (m)	$\lvert\theta_1-\theta_2\rvert = 2^{\circ}C$	$\lvert\theta_1-\theta_2\rvert = 6^{\circ}C$
500	45m	77m
750	55	95
1000	63	110
1500	77	134

TABLE A.2-2

THERMAL INTERNAL BOUNDARY LAYER DEPTH AFTER MITCHELL, EQN (A.2-2)

a. $(\Delta T/\Delta Z)_1 = +0.001^{\circ}C/m$; $(\Delta T/\Delta Z)_2 = -0.02^{\circ}C/m$

Inland Fetch (m)	$\lvert\theta_1-\theta_2\rvert = 2^{\circ}C$	$\lvert\theta_1-\theta_2\rvert = 6^{\circ}C$
500	22m	38m
750	27	46
1000	31	53
1500	38	65

b. $(\Delta T/\Delta Z)_1 = +0.005^{\circ}C/m$; $(\Delta T/\Delta Z)_2 = -0.02^{\circ}C/m$

Inland Fetch (m)	$\lvert\theta_1-\theta_2\rvert = 2^{\circ}C$	$\lvert\theta_1-\theta_2\rvert = 6^{\circ}C$
500	20m	35m
750	24	42
1000	28	49
1500	35	60

TABLE A.2-3

THERMAL INTERNAL BOUNDARY LAYER DEPTH AFTER TURNER, EQN (A.2-3)

Inland Fetch (m)	u = 3 m/s	u = 6 m/s
500	45m	32m
750	55	39
1000	63	45
1500	77	55

TABLE A.2-4

INTERNAL BOUNDARY LAYER DEPTH AFTER PLATE, EQNS. (A.2-4 & -5)

Inland Fetch (m)	Depth
500	78m
750	108
1000	136
1500	189

TABLE A.3-1

INTERNAL BOUNDARY LAYER DEPTH BASED ON AVERAGED COMPOSITE OF TABLES A.2-1 THROUGH A.2-4

Inland Fetch (m)	Mean Depth	Range
500	60m	20 to 173m
750	75m	24 to 212m
1000	88m	28 to 245m
1500	109m	35 to 300m

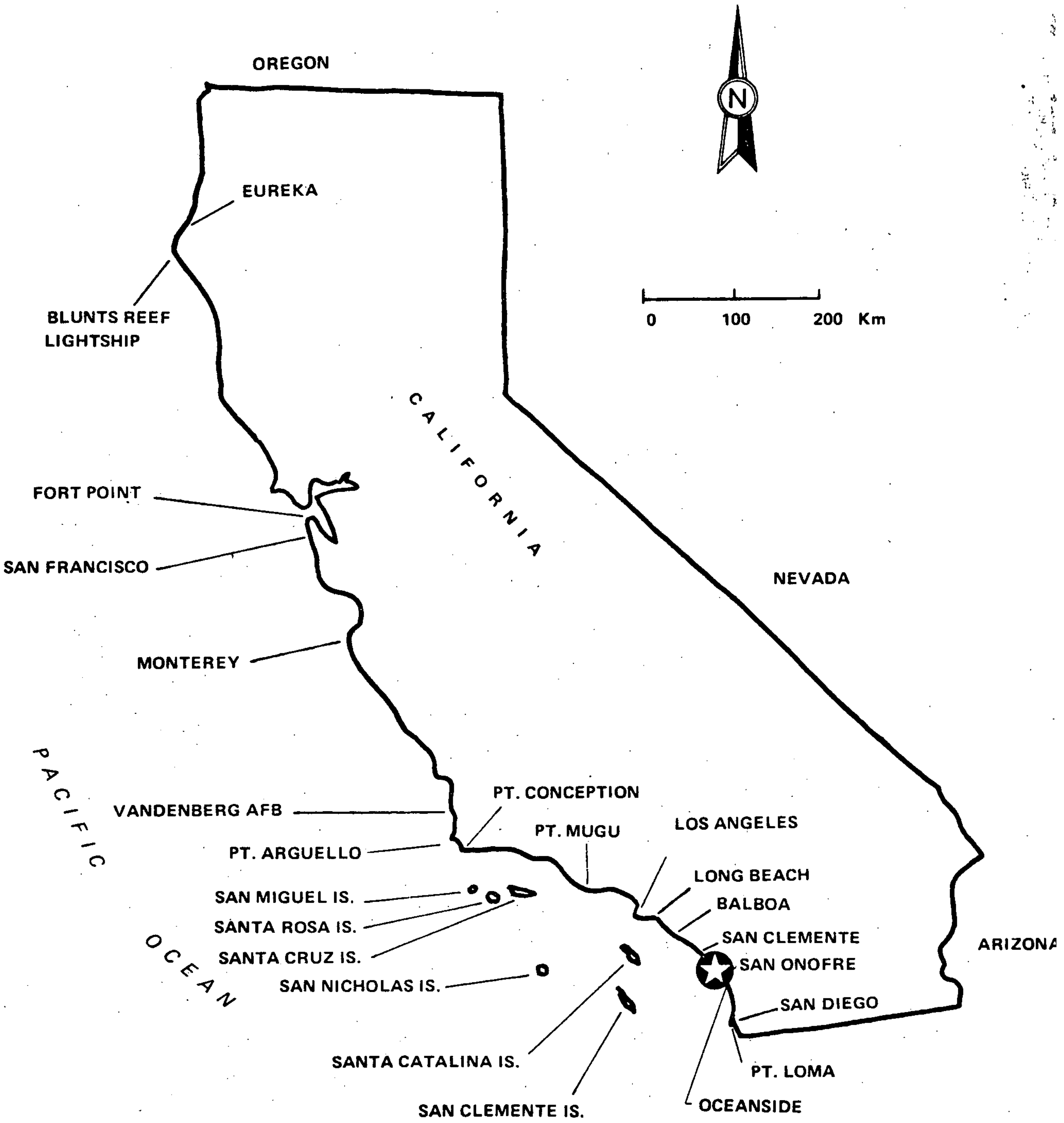

FIGURE A.3-1

MAP OF CALIFORNIA AND ADJACENT COASTAL WATERS

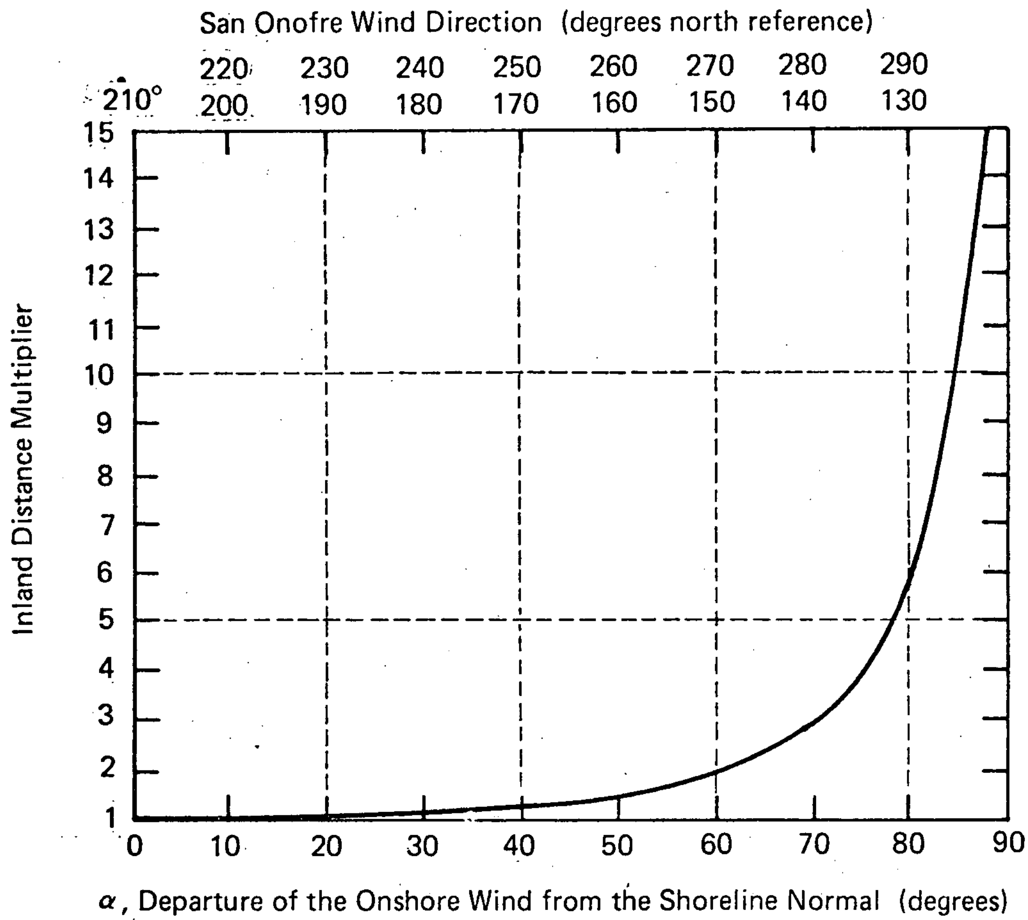

FIGURE A.3-2 INLAND DISTANCE MULTIPLIER AS A FUNCTION OF WIND ANGLE TO SHORELINE

TABLE A.3-1

APPROXIMATE SAN ONOFRE WIND DIRECTION
FREQUENCY AND INLAND DISTANCE MULTIPLIERS

This table is based on 1/75 to 1/76 10 m Bluff Tower Data (Septoff et al., 1967). Inland Distance Multiplier Times Inland Distance Equals Overland Fetch. The direction and total frequencies have been rounded.

Onshore Wind Direction	Wind Direction Frequency (%) of (All Directions)	Wind Direction Frequency (%) of (Onshore Only)	Inland Distance Multiplier
SE	7	12	4
SSE	7	12	2
S	6	10	1
SSW	6	10	1
SW	6	10	1
WSW	7	12	1
W	10	17	2
WNW	9	16	4
TOTAL	58	100	

INLAND TOWER SITING

IT IS UNLIKELY THAT A STABLE ONSHORE (NORMAL TO THE SHORELINE) FLOW WITH A DISTINCT TIBL WILL OCCUR, SO THAT A 40M TOWER LOCATED 600M OR MORE INLAND IS NOT LIKELY TO SPAN A TRANSITION ZONE OF A TIBL

AUG 2 7 1976

Docket Nos. 50-361
and 50-362

Karl Kniel, Chief, Light Water Reactors Branch No. 2, DPM

FORTHCOMING MEETING WITH SOUTHERN CALIFORNIA EDISON TO DISCUSS PROPOSED ONSHORE TRACER TESTS AT SAN ONOFRE

DATE & TIME:	Wednesday, September 8, 1976 1:00 p.m.
LOCATION:	Room P-110 Bethesda, Maryland
PURPOSE:	Technical Presentation by Edison of Proposed Tests.
PARTICIPANTS:	SOUTHERN CALIFORNIA EDISON (K. P. Baskin, D. Nunn, D. F. Pilmer, W. Moody, L. J. Brunton, et al) NRC - STAFF (E. Markee, J. Goll, R. Froelich, A. Burger, H. Rood, et al)

Original Signed by
Harry Rood

Harry Rood
Light Water Reactors
Branch No. 2
Division of Project Management

OFFICE ➤	DPM:LWR #2					
SURNAME ➤	HRood:mt					
DATE ➤	8/ /76					

Form AEC-93 (Rev. May 14, 1947) AECM 0240 | Note and return. | For signature. | For information.

TO (Name and unit)	INITIALS / DATE	REMARKS
J. Collins, ETSB, DSE THRU: F. Congel, RAB RLG for		
TO (Name and unit)	INITIALS / DATE	REMARKS Docket Nos. 50-206/361/362
TO (Name and unit)	INITIALS / DATE	REMARKS MINI-REVIEW FOR APPENDIX I SUBMITTAL FOR SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1

FROM (Name and unit) RLG

R. L. Gotchy, RIS
RAB/DSE
P-214

PHONE NO. 27955

DATE 8/25/76

REMARKS

The RAB staff has reviewed the subject submittal and found it adequate for the Appendix I dose assessment. This review was performed by R. Gotchy and took approximately 1.5 man-hours.

Reg

USE OTHER SIDE FOR ADDITIONAL REMARKS

GPO : 1971 O - 445-469

AUG 2 4 1976

DOCKET NOS. 50-206
50-361
and 50-362

APPLICANT: SOUTHERN CALIFORNIA EDISON COMPANY (SCE)

FACILITY: SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1, UNIT 2 AND UNIT 3

SUMMARY OF MEETING HELD ON JULY 26, 1976 CONCERNING SAN ONOFRE METEOROLOGY

On July 26, 1976, we met in Bethesda, Maryland with representatives of SCE, and their consultants to discuss the above subject. A list of the attendees at the meeting is attached.

The staff had previously evaluated data collected from the onsite meteorological tower. As a result of this evaluation (specifically relating to data collected between 1973 and 1975) the staff found the data to be of questionable value for calculating X/Q's for the region into which an effluent will be released at the San Onofre site. Specific areas of the staff's concern related to data recovery, instrument errors and the use of the tower data in describing the air layer into which plant effluents would normally flow. The licensee has continued to review their meteorological program and has performed additional meteorological evaluations during the last six months to respond to the staff's concerns. The licensee has also evaluated the installation of an additional tower at an inland location outside the San Onofre site, and the performance of onshore tracer tests as alternate efforts to address the staff concerns relating to the San Onofre meteorology. During the July 26, 1976 meeting the licensee presented the details of the additional meteorological analyses done and outlined the proposed onshore tracer tests. The attached copies of the licensee's handouts provide a summary of the licensee's presentations.

The licensee stated that he wants to proceed with the onshore tracer tests (in lieu of installing an additional permanent tower) and take credit for turbulence and wake effects. The licensee indicated that the tracer tests would be done as an accelerated program. The program description would be submitted formally to the NRC within 6-8 weeks and the results would be available in April 1977. The licensee asked for staff agreement with the approach and he informed us that the FSAR with

OFFICE ➤						
SURNAME ➤						
DATE ➤						

the operating license application for Units 2 and 3 was planned to be tendered in November 1976. This submittal would be made 38 months prior to the scheduled date of fuel loading for early staff review because, in the licensee's opinion, more than the usual time may be required for completing the review process.

We stated that we could not decide now whether onshore tracer studies are acceptable in lieu of data collected from an additional permanent tower. It may be desirable to see preliminary results of the tracer studies in case we see some adverse feature that we cannot anticipate. With regard to the November 1976 submittal of the FSAR for Units 2 and 3, we said that we were not at that time in a position to tell the licensee that the FSAR will pass the acceptance review or that the application for operating licenses will be docketed without the meteorological information.

The licensee confirmed that the continuing work to resolve the San Onofre meteorology will not affect the ongoing work on the Unit 1 sphere enclosure project. The Unit 1 containment leakage test will be performed to the more restrictive allowable leakages based on the atmospheric conditions described in Regulatory Guide 1.4.

At the conclusion of the meeting, SCE informed us that they need to get the NRC's position on these matters soon so that they can proceed with timely action. To this end the licensee will write a letter to the NRC within one to two weeks from the date of the meeting committing to perform the onshore tracer study program and to provide the program description within another six to eight weeks. Based on these commitments, the licensee will ask the NRC if these steps are sufficient for tendering the application in November 1976 in anticipation of its acceptance and docketing of the application for staff review.

A. Burger
Operating Reactors
Branch No. 1
Division of Operating Reactors

H. Rood
Light Water Reactors
Branch No. 2
Division of Project Management

Attachments:
As stated

OFFICE ➤	cc: See page 3	ORB #1	DPM:LWR #2			
SURNAME ➤		ABurger (mt)	HRood			
DATE ➤		8/ /76	8/ /76			

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

cc: Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas and Electric Company
ATTN: Mr. Jack E. Thomas
Vice President - Electric
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Rollin E. Woodbury, General Counsel
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Chickering & Gregory, General Counsel
San Diego Gas and Electric Company
111 Sutter Street
San Francisco, California 94104

Mr. David Sakai
845 North Perry Avenue
Montebello, California 90640

Frederick P. Sutherland, Esq.
Center for Law in the Public Interest
10203 Santa Monica Boulevard
Los Angeles, California 90067

Mr. Kenneth E. Carr
City Manager
City of San Clemente
100 Avenido Presidio
San Clemente, California 92672

Alan R. Watts, Esq.
Assistant City Attorney
City Hall
Anaheim California 92805

Lawrence Q. Garcia, Esq.
California Public Utilities Commission
5066 State Building
San Francisco, California 94102

George Spiegel, Esq.
2600 Virginia Avenue, N. W.
Washington, D. C. 20036

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTENDANCE LIST

MEETING WITH SOUTHERN CALIFORNIA EDISON COMPANY
SAN ONOFRE NUCLEAR GENERATING STATION
JULY 26, 1976

SAI

L. H. Teuscher

NUS

M. Septoff
J. H. Taylor

SDGE

W. D. Griffith

NRC - STAFF

A. Burger
R. W. Froelich
E. H. Markee
J. Goll

Dames & Moore

R. G. Allen

SCE

L. J. Brunton
K. P. Baskin
W. C. Moody
D. E. Nunn

NOAA

I. VanderHoven

OFFICE ➤						
SURNAME ➤						
DATE ➤						

AGENDA

San Onofre Meteorology
July 26, 1976

I. Introduction

II. Discussion of Additional Meteorological Analyses

   A. Analysis of Delta T Data From Other California Coastal Sites

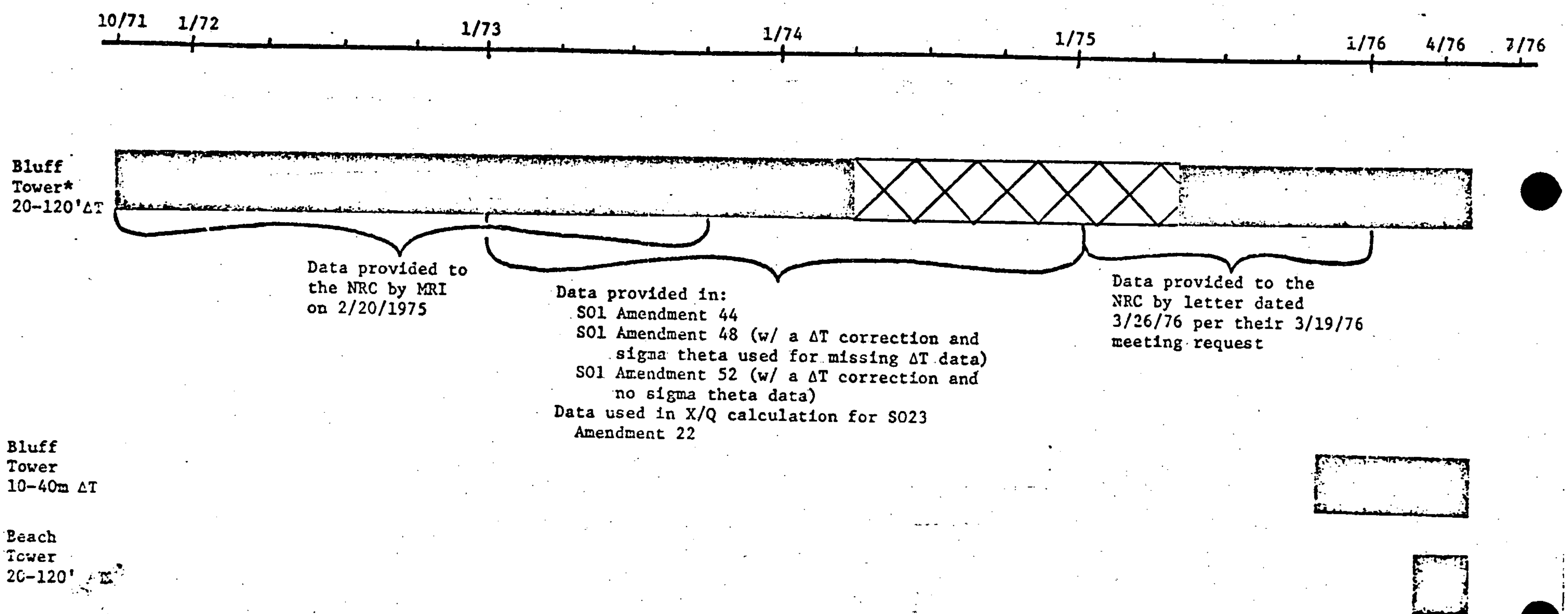
   B. Analysis of New Delta T Instrumentation Data

   C. Analysis of Bluff Tower Data Versus Beach Tower Data

III. Discussion of Proposed Onshore Tracer Tests

IV. Conclusions

SAN ONOFRE METEOROLOGICAL DATA COLLECTION



*Cross-hatched area indicates that period of data to which a delta T correction was applied in Amendments 48 and 52 and the 3/26/76 submittal.

SOUTHERN CALIFORNIA EDISON COMPANY

SAN ONOFRE NUCLEAR GENERATING STATION

Discussion of Additional Meteorological Analyses

July 26, 1976

INTRODUCTION

Additional meteorological work has been performed during the past six months to further evaluate the items of concern stated by the NRC staff relative to the San Onofre meteorological data measured to date. Specifically, the characteristics in question are the data's validity and whether or not the data reflect representative measurements of the region into which an effluent will be released at the San Onofre site.

I. California Coast Tower Comparison Study

A. Objective

Compare Pasquill Class frequency distributions from meteorological towers located on the California coast with exposures similar to San Onofre.

B. Method

Two years of delta temperature data from Diablo Canyon and one year of data from Point Conception (Figure 1) was classified by thermal gradient into Pasquill classes.

C. Results (Figure 2)

1. Comparing the three distributions (Table 1, Columns 7, 12, and 14) indicates that the extreme classes (A and G) increase with decreasing latitude.

2. Overall frequency distributions show similar shapes with the extreme measurements also apparently a function of size of the measurement interval (Figure 2).

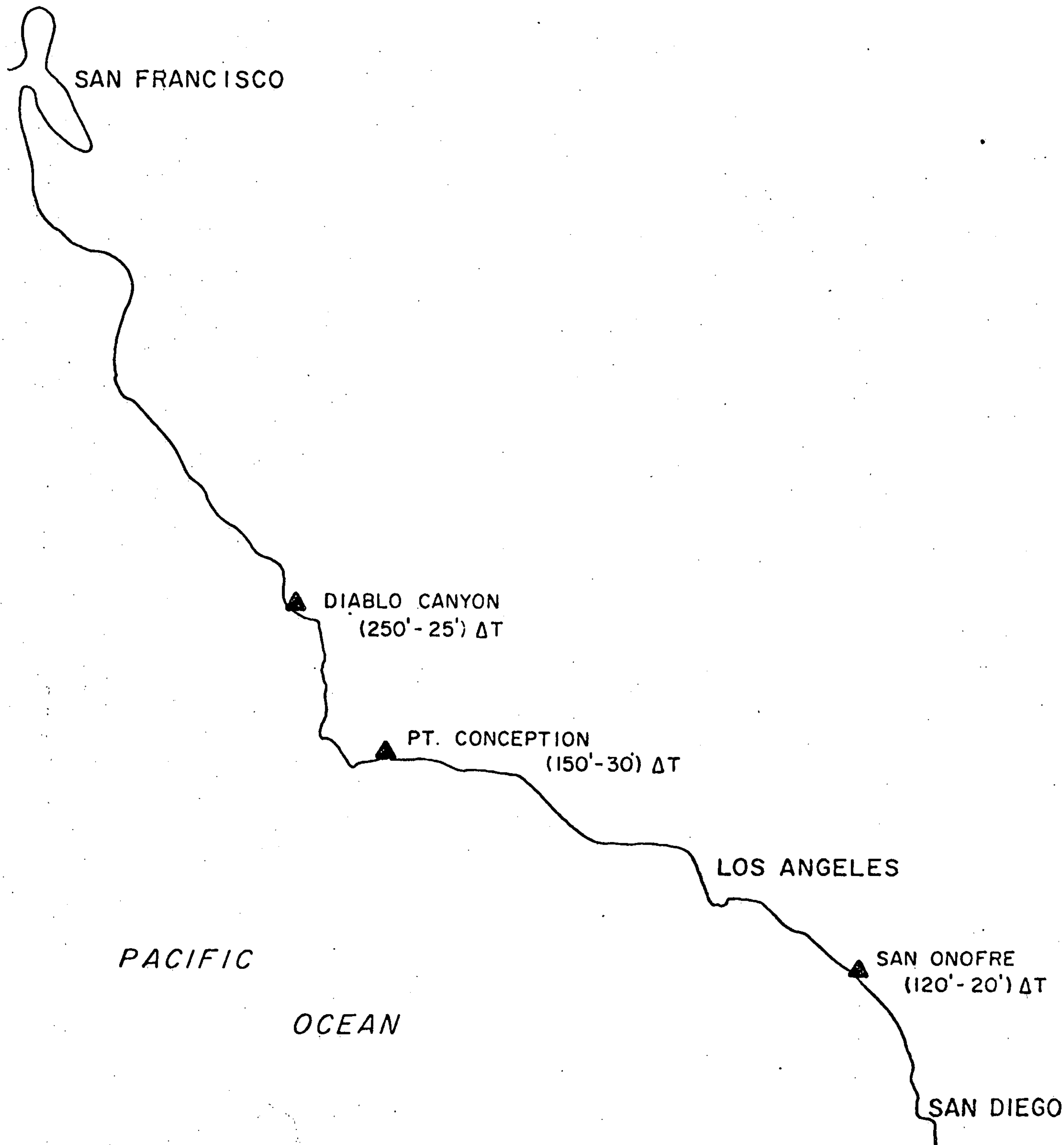

SAN FRANCISCO
DIABLO CANYON
(250'-25') ΔT
PT. CONCEPTION
(150'-30') ΔT
LOS ANGELES
SAN ONOFRE
(120'-20') ΔT
SAN DIEGO
PACIFIC
OCEAN

FIGURE 1

Table 1

COMPARISON OF STABILITY CATEGORIES

	1	2	3	4	5	6		7	8	9	10	11	12	13
	Amendment 44 4/17/75	Amendment 48 8/20/75	Amendment 52 12/3/75	New Instrumentation Installed 5/75	New Instrumentation Installed 5/75	New Instrumentation Installed 10/75		Includes All Data	Bluff Tower	Bluff Tower	Beach Tower	Beach Tower	Diablo Canyon	Point Conception
Collection Period:	1/25/73-1/24/75	1/25/73-1/24/75	1/25/73-1/24/75	1/25/75-1/29/76	11/75-4/76	11/75-4/76		1/25/73-1/24/76	3/76-4/76	3/76-4/76	3/76-4/76	3/76-4/76	5/73-4/75	3/71-4/72
ΔT HEIGHTS:	20-120'	20-120'	20-120'	20-120'	20-120'	10-40m (Reg. Guide)		20-120'	20-120'	20-120'	20-120'	20-120'	25-250'	30-150'
DIRECTIONS:	ALL	ALL	ALL	ALL	ALL	ALL		ALL	ALL	ONSHORE	ALL	ONSHORE	ALL	ONSHORE
NOTES:		Data contains ΔT correction. Sigma theta mistakenly utilized.	Data contains ΔT correction. Sigma theta removed.	Provided by letter dated 3/26/76. Data includes new ΔT sensor installed 5/75.				Data contains ΔT correction. Sigma theta removed.						
STABILITY CATEGORY						Primary	Secondary							
A	30.72	21.59	23.00	32	20.6	25.7	25.6	27.0	28	48	35	54	3.0	26.9
B	4.18	2.40	2.54	4										
C	3.64	3.23	3.43	5	3.0	11.5	7.0	7.1	4	6	11	15	5	7.9
D	15.09	20.83	21.93	20	18.9	14.6	19.2	21.7	23	33	18	22	43	32.4
E	19.90	21.22	21.14	13	16.3	16.9	16.7	17.8	14	10	10	6	38	21.1
F	8.65	10.55	9.16	8	13.4	17.0	17.2	8.4	12	3	10	2	8	7.7
G	17.81	20.18	18.80	18	27.9	14.4	14.2	18.3	20	0	16	1	1	4.0

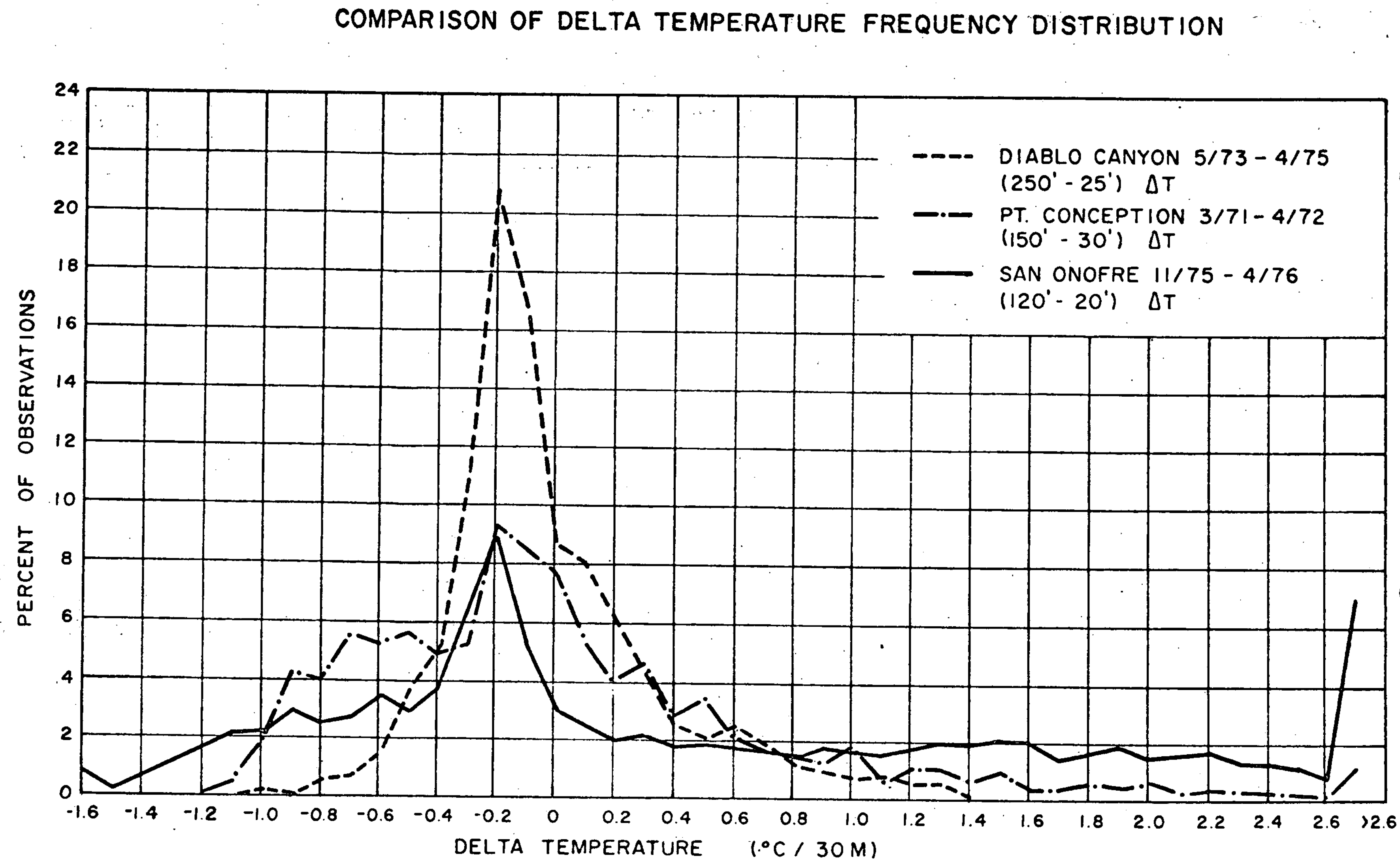

COMPARISON OF DELTA TEMPERATURE FREQUENCY DISTRIBUTION
DIABLO CANYON 5/73 - 4/75 (250' - 25') ΔT
PT. CONCEPTION 3/71 - 4/72 (150' - 30') ΔT
SAN ONOFRE 11/75 - 4/76 (120' - 20') ΔT
PERCENT OF OBSERVATIONS
24
22
20
18
16
14
12
10
8
6
4
2
0
-1.6
-1.4
-1.2
-1.0
-0.8
-0.6
-0.4
-0.2
0
0.2
0.4
0.6
0.8
1.0
1.2
1.4
1.6
1.8
2.0
2.2
2.4
2.6
>2.6
DELTA TEMPERATURE (°C / 30 M)

FIGURE 2

II. Delta-T Comparison Study

A. Objectives

1. Compare 6 months of data (November 1975-April 1976) taken on the Bluff tower at the height of the original instrumentation (120'-20') with concurrent measurements at (40m-10m).

2. Compare delta-temperature measurements taken from two separate instrument systems at (40m-10m) for congruity.

3. Compare these data with data taken for previous years.

B. Method - Develop Pasquill class frequency distributions for both measurement intervals.

C. Results

1. Pasquill "A" frequency increased from 20.6% to 25.7% for the (120'-20') ΔT to the (40m-10m) ΔT, respectively. (Table 1, Columns 5 and 6.)

2. Pasquill "G" frequency decreased from 27.9% to 14.4% for the (120'-20') ΔT to the (40m-10m) ΔT, respectively. (Table 1, columns 5 and 6.)

3. The two separate instrument systems measuring the delta temperature interval (40m-10m) compare closely, especially in the extreme categories - A and G. (Table 1, columns 6 primary and secondary.)

4. If χ/Q calculations were performed with these data, the (120'-20') ΔT would be the most conservative. (Table 1, columns 5 and 6.)

5. Pasquill class distributions are consistent with prior years. (Table 1, columns 1, 2, 3, and 4.)

III. Bluff-Beach Tower Comparison Study

A. Objectives

1. Identify and quantify the postulated "bluff-effect" which is suspected of modifying the parameters measured on the San Onofre Bluff tower.

2. Compare the diffusion climatology for onshore winds at both the Beach and Bluff towers for conservatism.

B. Method

1. Linear regression analyses on concurrently measured wind speeds and delta temperatures were performed for stable, unstable and onshore, offshore cases of Bluff tower stability and wind direction.

2. Frequency distributions of relative concentrations (X/Q) for both the Beach and the Bluff towers were calculated from the same data base at each tower.

C. Results

1. 30 ft. Beach wind speed versus 10 meter Bluff wind speed for onshore winds shows an average increase at the Bluff of 0.6 mph for both unstable and stable cases.

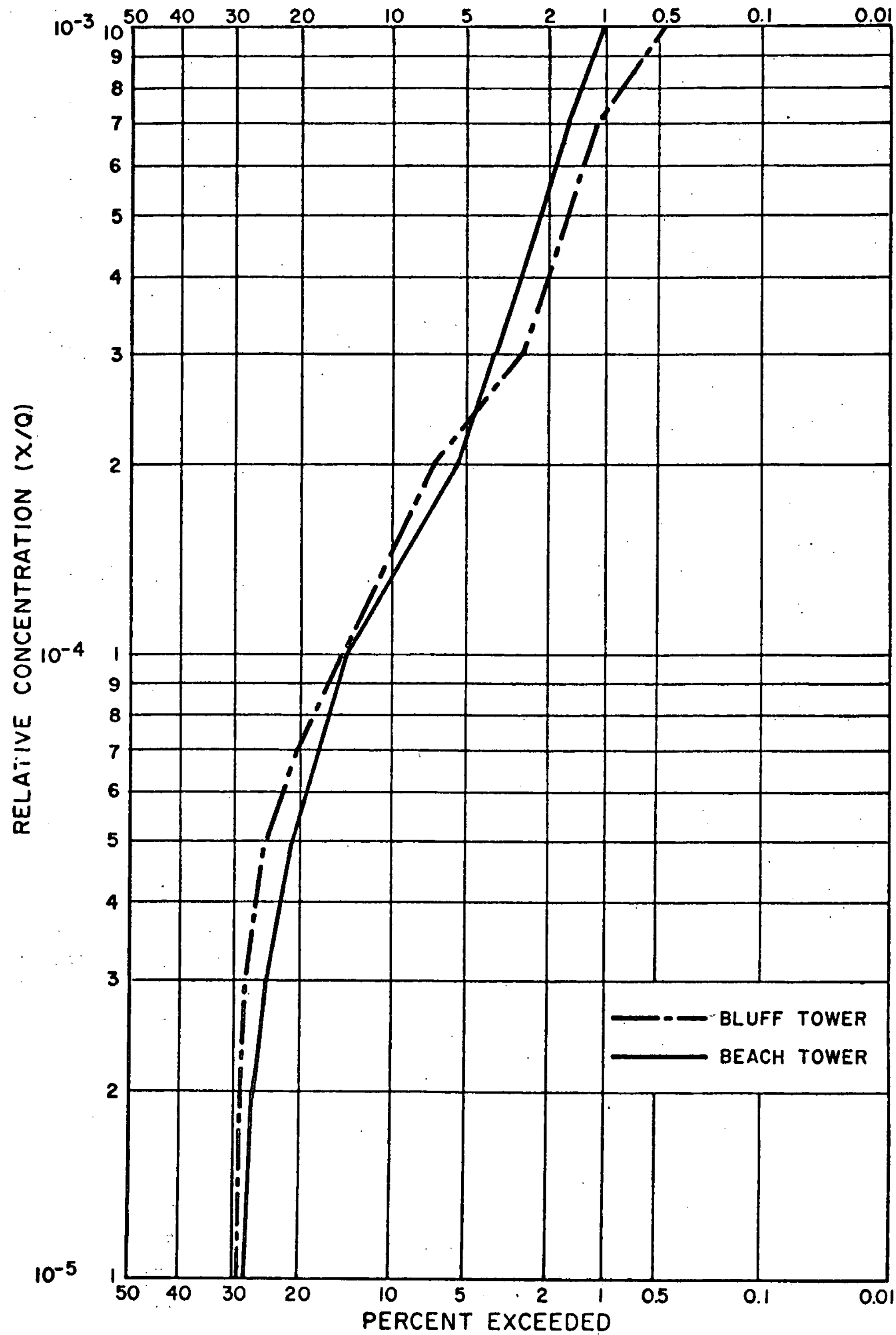
2. 30 ft. Beach wind speed versus 10 meter Bluff wind speed for offshore winds shows an average decrease at the Beach of 2.9 mph for stable cases. Unstable offshore cases occur less than 1.0% of time.

3. For the purpose of defining the onshore wind direction diffusion climatology at SONGS, the Bluff tower measurements give similarly conservative values as the Beach Tower (Figure 3).

4. Atmospheric stabilities measured concurrently on each tower indicate a tendency toward more stable conditions on the Bluff (Table 1, Columns 8, 9, 10, and 11).

5. For strongly unstable cases, onshore flow occurs 94% to 99% of the time. For very stable conditions, offshore winds prevail 96% to 99% of the time (Table 2).

6. Indications are that the "bluff effect" is not causing the high incidence of Pasquill "A" at San Onofre.



SAN ONOFRE NUCLEAR GENERATING STATION
ONSHORE X/Q (HOURLY)
CUMULATIVE FREQUENCY DISTRIBUTION
DISTANCE = 576 METERS
CA = 0
PERIOD: 3/1 - 4/30/76

FIGURE 3

TABLE 2

Stability Class	% Onshore		% Offshore	
	Bluff	Beach	Bluff	Beach
A	98.9	94.0	1.1	6.0
B & C	90.0	85.6	10.0	14.4
D	81.0	71.9	19.0	28.1
E	44.6	34.2	55.4	65.8
F	11.2	11.9	88.8	88.1
G	1.2	3.8	98.8	96.2
All	57.7	60.1	42.3	39.9

Period: (3/1-4/30/76) .

Bluff Tower: (120'-20') ΔT, 10 m Wind

Beach Tower: (120'-20') ΔT, 30 ft. wind

CONCLUSIONS

1. The present San Onofre Bluff Tower is giving valid measurements for that location. This is true because of intensive maintenance and calibration practices now being performed; and, a demonstrated consistency between redundant instruments.

2. The data now being measured on the San Onofre meteorological instrument systems are consistent with data measured in prior years. The high incidence of atmospheric instability is real at that site, being evident at locations above and below the coastal bluff.

3. The data taken from the present Bluff Tower does show evidence of a bluff-effect in that the lower level winds are apparently increased or decreased, depending on wind direction. However, the measured shift in stability towards more stable onshore winds does compensate for the increased wind speeds. The overall effect (for the data period analyzed) on the diffusion parameters calculated from these data is to show that the Bluff Tower gives equally conservative values as the Beach Tower for onshore winds.

4. Past data measured on the San Onofre tower with the (120'-20') ΔT system is conservative when compared to data collected at elevations specified in Regulatory Guide 1.23 (40m-10m) ΔT .

OBJECTIVES

- PRIMARY

  CHARACTERIZE ATMOSPHERIC DISPERSION TO PERMIT REALISTIC CALCULATIONS OF SHORT TERM ACCIDENT DISPERSION FACTORS

- SECONDARY

  INVESTIGATE DISPERSION UNDER LESS RESTRICTIVE CONDITIONS TO PERMIT EVALUATION OF ROUTINE RELEASES

TRACER RELEASE

- SAME AS BEACH TRACER TESTS
- ELEVATIONS APPROXIMATELY AT GROUND LEVEL BELOW BLUFF AND TOP OF DOME
- SMOKE AT RELEASE POINT
- UNIT 1 AND UNITS 2 OR 3

SAMPLE COLLECTION

- AUTOMATIC AND SEMI-AUTOMATIC BAG SAMPLES
- TWO SETS OF SAMPLING ARCS
- 700 METERS AND 300 METERS
- VERTICAL SAMPLING

SAMPLE ANALYSIS

- SAME AS BEACH TOWER TESTS

METEOROLOGICAL MEASUREMENTS

- TEMPORARY TOWER WITH TWO LEVELS OF INSTRUMENTATION
- 2 TO 3 ADDITIONAL 10 METER MASTS
- BLUFF TOWER
- BEACH INSTRUMENTATION

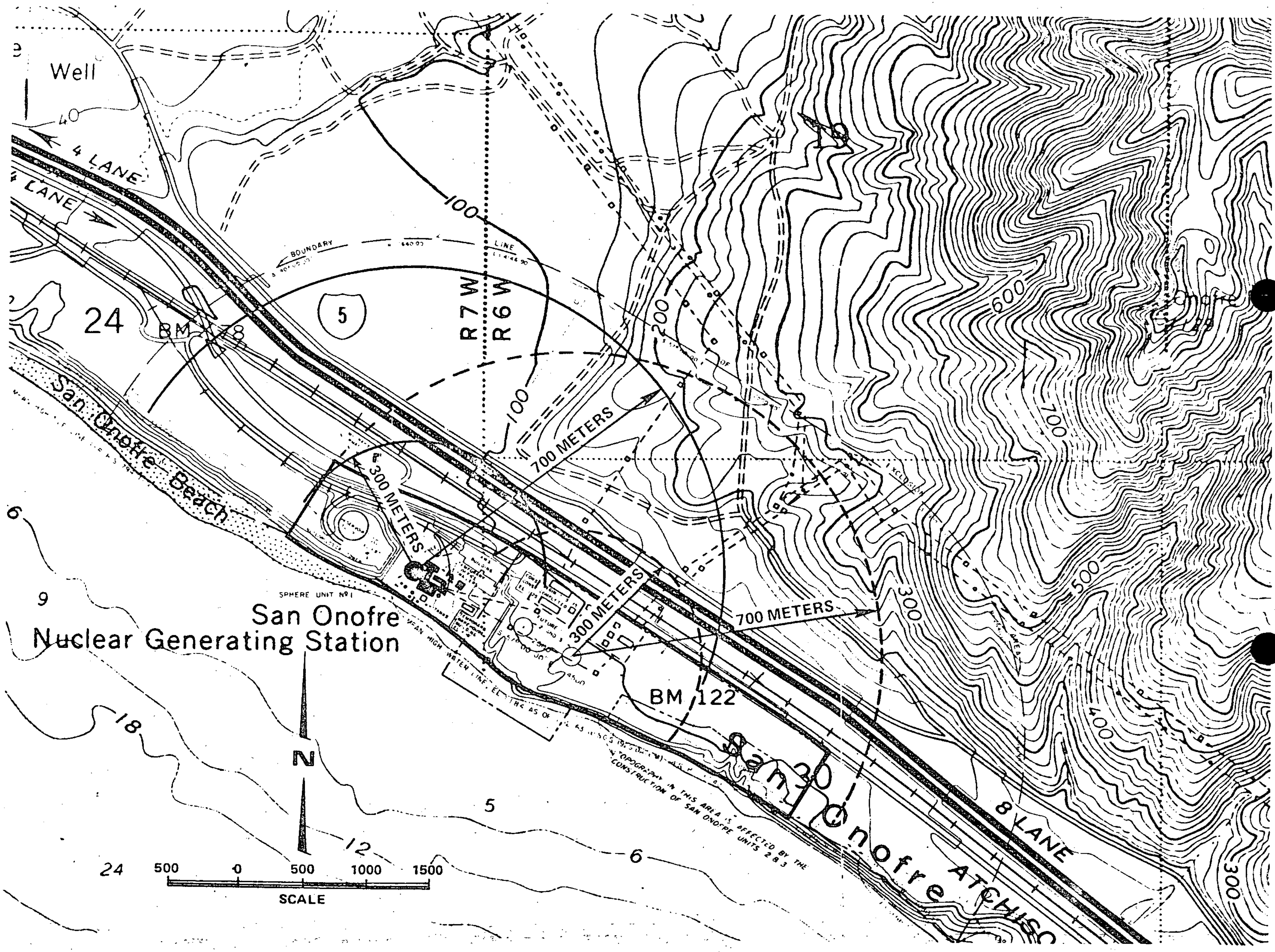
OPERATIONAL PLAN

- METEOROLOGICAL CONDITIONS
  - LIGHT TO MODERATE WINDS ($\leq$ 8 MPH)
  - PRIMARILY NEUTRAL TO SLIGHTLY STABLE CONDITIONS; SOME UNSTABLE

- ONE HOUR SAMPLING PERIODS

DATA ANALYSES

- SIMILAR TO BEACH TRACER TEST PROGRAM (NUS-1702 INTERIM)



Well
4 LANE
4 LANE
BOUNDARY
LINE
24
5
R 7 W
R 6 W
100
200
600
700
500
400
300
700 METERS
300 METERS
300 METERS
700 METERS
San Onofre Beach
SPHERE UNIT Nº1
San Onofre
Nuclear Generating Station
BM 122
San Onofre
8 LANE
ATCHISO
TOPOGRAPHY IN THIS AREA IS AFFECTED BY THE
CONSTRUCTION OF SAN ONOFRE UNITS 2 & 3
N
6
9
18
5
12
6
24
500
0
500
1000
1500
SCALE

JUL 2 2 1976

Docket Nos. 50-206
50-361
and 50-362

Karl Kniel, Chief, Light Water Reactors Branch No. 2, DPM

FORTHCOMING MEETING WITH SOUTHERN CALIFORNIA EDISON TO DISCUSS SAN ONOFRE METEOROLOGY

DATE & TIME: Monday, July 26, 1976 - 1:00 p.m.

LOCATION: Room P-114
Bethesda, Maryland

PURPOSE: Technical Presentation by Edison
(see attached agenda)

PARTICIPANTS:

SOUTHERN CALIFORNIA EDISON
(K. P. Baskin, D. Nunn, D. F. Pilmer, W. Moody, L. J. Brunton, et al)

NUCLEAR UTILITY SERVICES
(J. Taylor, M. Septoff, et al)

DAMES & MOORE
(R. Allen, et al)

NRC - STAFF
(E. Markee, J. Goll, R. Froelich, A. Burger, H. Rood, et al)

Original Signed by
Harry Rood

Harry Rood
Light Water Reactors
Branch No. 2
Division of Project Management

Attachment:
Agenda

OFFICE ➤	DPM:LWR #2					
SURNAME ➤	HRood:mt					
DATE ➤	7/ /76					

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

JUL 2 2 1976

Agenda

San Onofre Meteorology
July 26, 1976

I. Introduction

II. Discussion of Additional Meteorological Analyses

   A. Analysis of New Delta T Instrumentation Data (January 1975 - January 1976; 20-120')

   B. Analysis of New Delta T Instrumentation Data (November 1975 - April 1976; 10-40m)

   C. Analysis of Bluff Tower Data Versus Beach Tower Data (March 1976 - April 1976; 20-120')

   D. Analysis of Delta T Data From Other California Coastal Sites

III. Discussion of Proposed Addition; Meteorological Investigation Alternatives

   A. Installation of Additional Tower at Inland Site

   B. Performance of Onshore Tracer Tests

IV. Conclusions

50-362

May 18, 1976

Robert A. Purple, Chief, Operating Reactors Branch #1, DOR

BOMB THREAT TO SAN ONOFRE NUCLEAR GENERATING STATION

On May 11, 1976, the licensee provided us (by phone at 1:45 p.m.) the following fragmentary information.

At 6:30 a.m. Pacific Daylight Time, the Sheriff received a call concerning a bomb threat. The Sheriff promptly notified the San Onofre Unit No. 1 shift supervisor that he had been called by phone that a bomb will go off at 8:00 a.m. in the "Bechtel plant." (It is not yet clear whether the threat was directed at San Onofre Unit No. 1 or San Onofre Unit Nos. 2 and 3.) The plant security guard did not allow the day shift on site (San Onofre Unit Nos. 1, 2, and 3) and they evacuated the night shift. The reactor shift crew remained on duty and operation of San Onofre Unit No. 1 continued. The Marine Corps was notified and dispatched a contingent of Marines to the site. The security guard conducted a search of the site and did not find anything unusual. By 9:15 a.m. construction workers were allowed on site. Established security measures (to be taken in case of a bomb threat) remain in effect. The licensee will notify Region V.

/S/

Alfred Burger, Project Manager
Operating Reactors Branch #1
Division of Operating Reactors

cc: B. C. Rusche
E. G. Case
V. Stello
R. E. Heineman
R. S. Boyd
H. R. Denton
K. R. Goller
D. G. Eisenhut

DISTRIBUTION
Docket Files (50-206, ~~361~~, 362)
NRC PDRs
Local PRDs
ABurger
SMSheppard

OFFICE ➤	DOR:ORB#1					
SURNAME ➤	ABurger:dc /S/					
DATE ➤	5/18/76					

Docket

MAY 1 0 1976

DOCKET NOS. 50-361 and 50-362

FACILITY: San Onofre Units 2 and 3

APPLICANTS: Southern California Edison Company (SCE) and San Diego Gas and Electric Company (SDGE)

SUMMARY OF MEETING TO DISCUSS SCE RESPONSE TO THE STAFF'S 3/31/76 REQUEST FOR ADDITIONAL INFORMATION

On April 16, 1976, the NRC staff met with the applicant's representatives in Bethesda, Maryland to discuss the above subject. Attendees at the meeting are given in Enclosure 1. Three general areas were discussed at the meeting. These were (1) the recently conducted tracer tests (2) an assessment of evacuation doses (3) beach use and beach configuration.

With regard to the recent tracer tests, the applicants indicated that some of the information submitted in the April 12, 1976 response to the March 31, 1976 questions was in error, and that corrected replacement pages to the tracer report (NUS-1702) would be provided.*

Original Signed by
Harry Rood

Harry Rood
Light Water Reactors
Branch No. 3
Division of Project Management

Enclosure:
Attendance list

*Corrected pages were docketed on May 6, 1976.

OFFICE➤	DPM:LWR #3					
SURNAME➤	HRood:mjf HR					
DATE➤	5/7/76					

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Docket

MAY 10 1976

ENCLOSURE 1
Attendees
San Onofre 2 & 3 Exclusion Area
April 16, 1976

Southern California Edison

D. Nunn
D. Pilmer
W. Moody
L. Brunton
K. Baskin
L. Teuscher

San Diego Gas & Electric

W. Griffith

Meteorology Res. Inc.

T. Lockhart

NUS

M. Goldman
M. Septoff
J. Taylor

NOAA

I. VanderHoven

NRC

H. Rood
R. Ross
J. McGurren
E. Markee
C. Ferrell
J. Goll
E. Hawkins
J. Sears
A. Burger

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Docket

APR 1 6 1976

DOCKET NOS. 50-361 and 50-362

FACILITY: San Onofre Units 2 and 3

APPLICANTS: Southern California Edison Company (SCE) and San Diego Gas and Electric Company (SDGE)

SUMMARY OF MEETING TO DISCUSS NUS TRACER TESTS

On March 19, 1976, the NRC staff met with the applicants' representatives to discuss the above subject. Attendees are given in Enclosure 1. Information presented by the applicants at the meeting is given in Enclosure 2. At the conclusion of the meeting it was agreed that the NRC staff would formally request additional information regarding the tracer tests and other items related to the upcoming hearings about a week after the meeting, and that the applicants would provide a response to the NRC request within about four weeks of the meeting.

Original Signed by
Harry Rood

Harry Rood
Light Water Reactors
Branch No. 3
Division of Project Management

Attachment:
Agenda

OFFICE ➤	DPM:LWR #3					
SURNAME ➤	HRood:mjf			P		
DATE ➤	4/16/76					

Form AEC-318 (Rev. 9-53) AECM 0240

APR 1 6 1976

Enclosure 1
Attendance List
San Onofre 2/3 Meeting
March 19, 1976

SOUTHERN CALIFORNIA EDISON CO.

K. Baskin
D. Pilmer
D. Nunn
E. Donovan
J. Brunton

SAN DIEGO GAS & ELECTRIC CO.

W. Griffith

SCIENCE APPLICATIONS INC.

L. Teuscher

NOAA

I. Vanderhoven

NUS

M. Septoff
J. Taylor
M. Goldman

BECHTEL

A. Nakashima
J. Hosmer

MRI

N. Hallanger
T. Lockhart

NRC

H. Rood
J. Goll
E. Markee
A. Burger
C. Ferrell
L. Soffer
D. Bunch
S. Treby
D. Tibbitts
R. Abbey

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ENCLOSURE 2

AGENDA

March 19, 1976

SUBJECT: Tracer Dispersion Studies and Evacuation Doses
San Onofre Nuclear Generating Station

I.	Introduction	SCE (K. P. Baskin)
II.	Tracer Study	NUS (M. Septoff)
	A. Objectives	
	B. Program Description	
	C. Results and Conclusions	
	D. Plans for Completion of Study	
III.	Utilization of Existing Bluff Tower Data	MRI (N. L. Hallanger)
IV.	Application of Tracer Study and Bluff Tower Data	NUS (John Taylor)
V.	Method of Calculating Evacuation Doses	NUS (M. I. Goldman)
VI.	Concluding Remarks, Schedule and Future Work	SCE (K. P. Baskin)

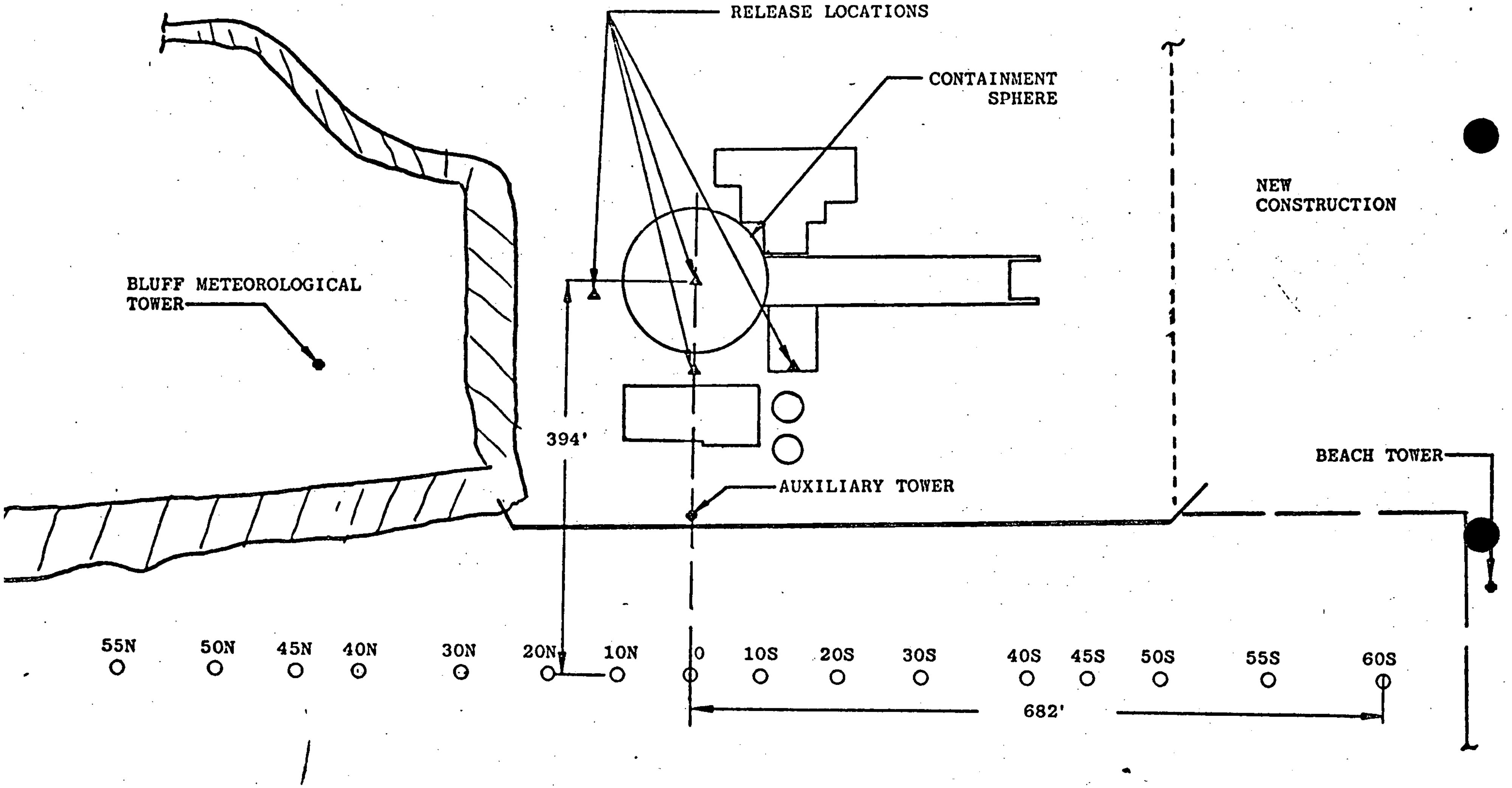
Preliminary Averages,[1] Songs Data Runs 2-21

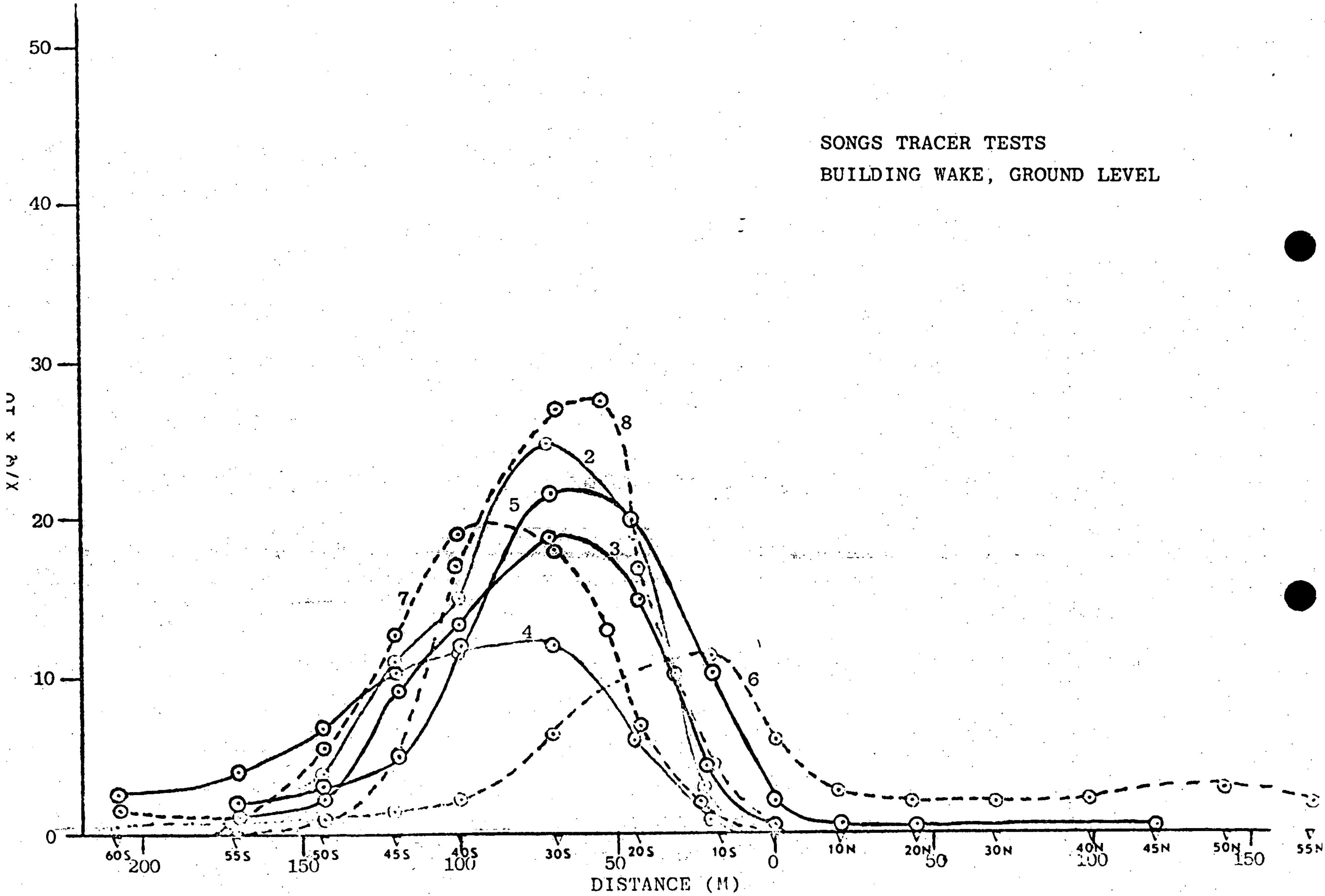
Run	Bluff WD	Bluff WS	Bluff Stability Class	Beach, Main Tower WD	Beach, Main Tower WS	Beach, Main Tower Stability Class	Beach, Auxiliary Tow[er] WD	Beach, Auxiliary Tow[er] WS
2	25°	12.5	G	15°	6.5 mph	G	30°	7.5 mph
3	25°	12.5	G	20°	7.0	G	30°	7.5
4	15°	12.5	G	0°	6.0	G	25°	7.5
5	45°	6.0	G	55° 3)	3.5	G	60°	5.0
6	70°	7.5	G	NA 2)	3.0	G	100° 3)	4.5
7	25°	9.5	G	5°	5.0	G	25°	6.5
8	25°	10.0	G	5°	5.0	G	20°	6.5
9	25°	8.0	G	10°	3.5	G	20°	6.0
10	25°	7.0	F	10°	3.5	G	20°	5.0
11	35°	7.0	G	30°	3.5	G	30°	5.0
12	25°	11.0	G	5°	6.0	G	25°	7.0
13	50° 3)	10.5	G	70° 3)	5.5	G	5° 3)	6.0
14	25°	12.0	G	345° 3)	5.5	F	25° 3)	7.5
15	30°	16.0	G	25°	11.0	G	35°	11.0
16	20°	15.0	F	15°	10.5	G	30°	8.5
17	90°	8.0	G	10°	3.0	G	20°	5.5
18	315°	9.5	G	355°	5.0	G	15°	7.0
19	5°	13.5	G	0°	8.5	G	25°	7.5
20	25°	12.0	G	10°	5.0	G	30°	8.5
21	0°	6.5	G	355°	4.5	G	20°	5.5

(notes-cont.)

NOTES:

1) AVERAGE WD AT MAIN BEACH TOWER, RUNS 1-6, AND AVERAGE WD AT AUXILIARY BEACH TOWER, RUNS 1-10, WERE ESTIMATED FROM 2-SECOND SAMPLES. ALL OTHER AVERAGES WERE CALCULATED.
2) WIND DIRECTION VARIED AROUND THE COMPASS - AVERAGE IS MEANINGLESS.
3) WIND SHIFT DURING DATA RUN.


RELEASE LOCATIONS
CONTAINMENT SPHERE
NEW CONSTRUCTION
BLUFF METEOROLOGICAL TOWER
394'
BEACH TOWER
AUXILIARY TOWER
55N
50N
45N
40N
30N
20N
10N
0
10S
20S
30S
40S
45S
50S
55S
60S
682'


SONGS TRACER TESTS
BUILDING WAKE, GROUND LEVEL
50
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7
4
6
60S 55S 50S 45S 40S 30S 20S 10S 0 10N 20N 30N 40N 45N 50N 55N
200 150 100 50 0 50 100 150
DISTANCE (M)

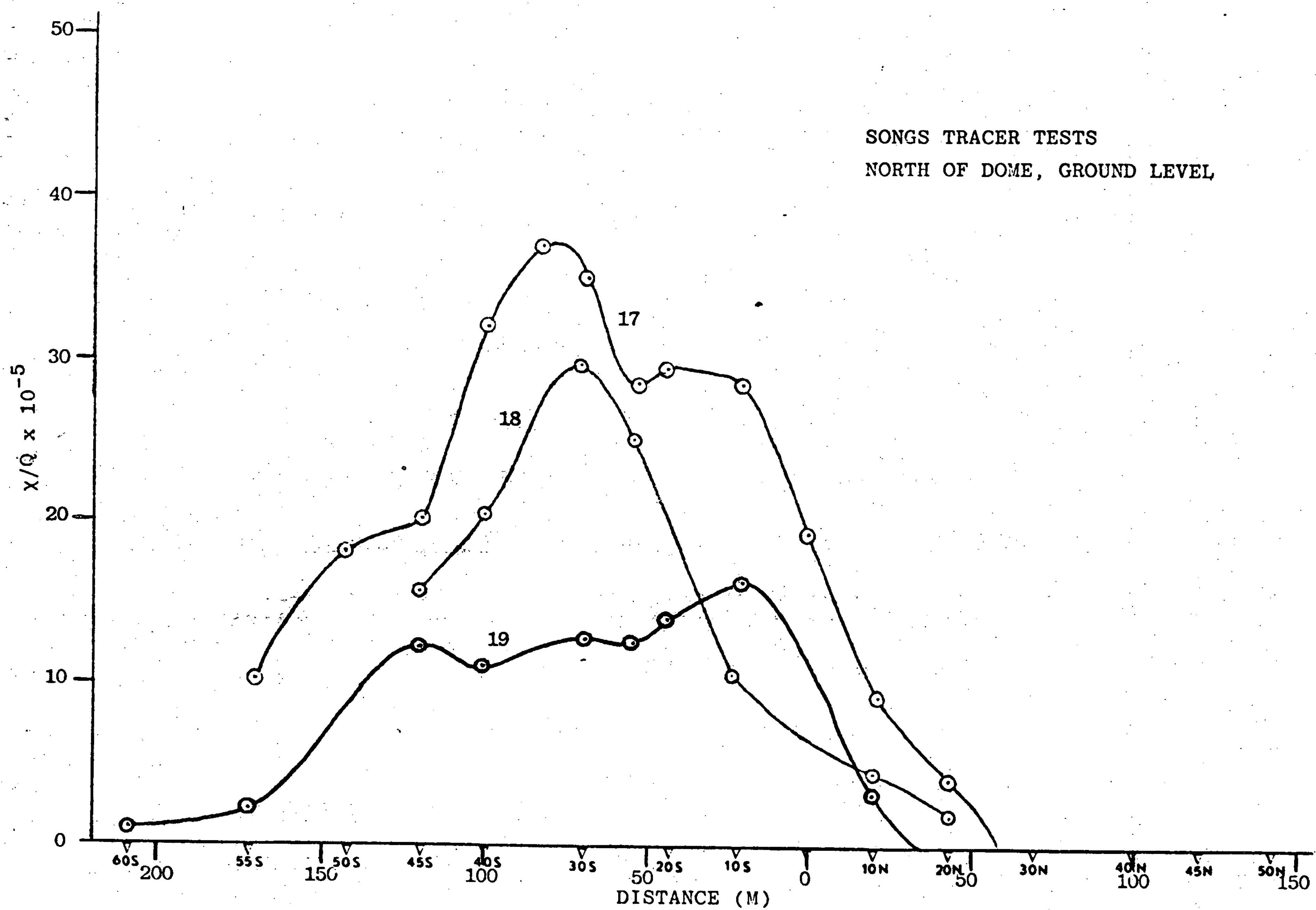
COMPARISON

OF CALCULATED AND MEASURED

X/Q

Ground Release - Seawall side of Containment

Test No.	Measured	Bluff Tower calculated	Bluff Tower red. factor	Beach Tower calculated	Beach Tower red. factor
2	2.5×10^{-4}	3.8×10^{-3}	15	7.4×10^{-3}	30
3	1.9×10^{-4}	3.8×10^{-3}	20	6.8×10^{-3}	36
4	1.2×10^{-4}	3.8×10^{-3}	32	8.0×10^{-3}	67
5	2.2×10^{-4}	8.0×10^{-3}	36	1.4×10^{-2}	64
6	1.1×10^{-4}	9.0×10^{-3}	82	2.3×10^{-2}	209
7	2.0×10^{-4}	4.2×10^{-3}	21	8.0×10^{-3}	40
8	2.8×10^{-4}	4.8×10^{-3}	17	9.6×10^{-3}	34

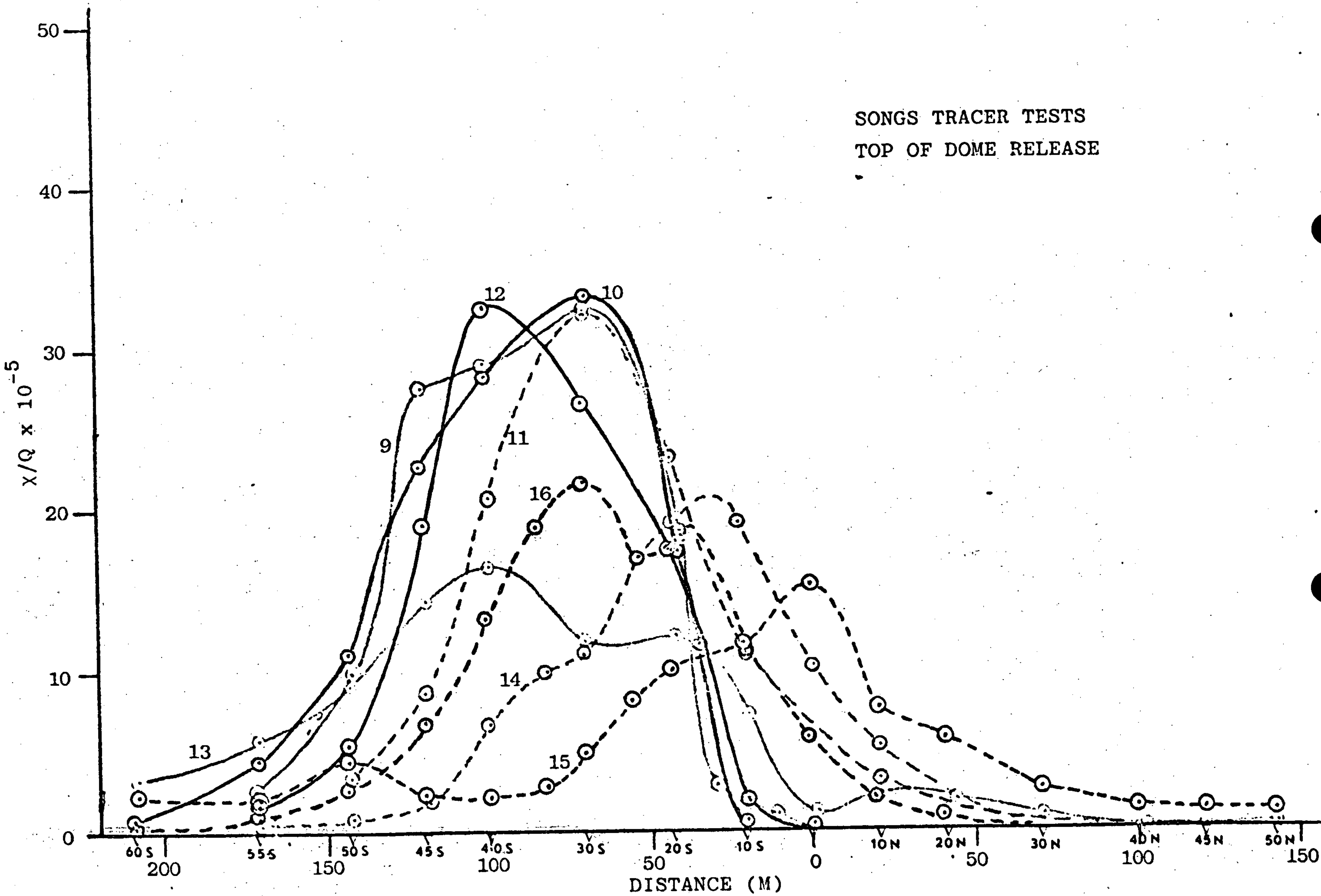


SONGS TRACER TESTS
NORTH OF DOME, GROUND LEVEL
χ/Q x 10^-5
50
40
30
20
10
0
17
18
19
60S
55S
50S
45S
40S
30S
20S
10S
10N
20N
30N
40N
45N
50N
200
150
100
50
0
50
100
150
DISTANCE (M)

COMPARISON
OF CALCULATED AND MEASURED
X/Q

Ground Release - North side of Containment

Test No.	Measured	Bluff Tower calculated	Bluff Tower red. factor	Beach Tower calculated	Beach Tower red. factor
17	3.7×10^{-4}	3.5×10^{-3}	9	9.4×10^{-3}	25
18	2.9×10^{-4}	3.0×10^{-3}	10	5.7×10^{-3}	20
19	1.6×10^{-4}	3.2×10^{-3}	20	5.1×10^{-3}	32



SONGS TRACER TESTS
TOP OF DOME RELEASE
χ/Q x 10^-5
50
40
30
20
10
0
12
10
9
11
16
14
13
15
60 S
55 S
50 S
45 S
40 S
30 S
20 S
10 S
0
10 N
20 N
30 N
40 N
45 N
50 N
200
150
100
50
0
50
100
150
DISTANCE (M)

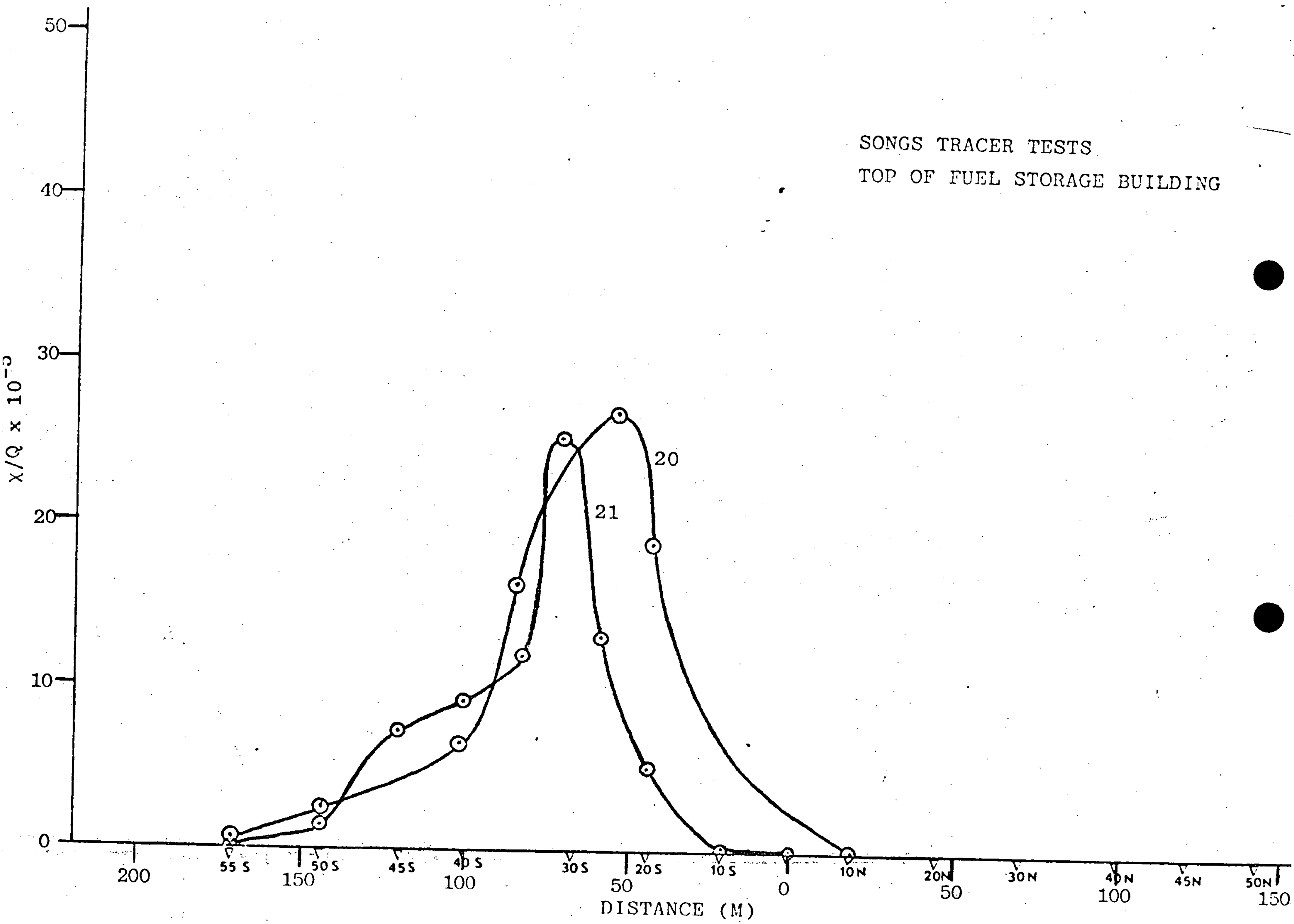
COMPARISON

OF CALCULATED AND MEASURED

X / Q

Top of Dome Release

Test No.	Measured	Bluff Tower		Beach Tower	
		calculated	red. factor	calculated	red. factor
9	3.2×10^{-4}	4.2×10^{-3}	13	9.6×10^{-3}	30
10	3.3×10^{-4}	1.9×10^{-3}	6	9.6×10^{-3}	29
11	3.2×10^{-4}	4.8×10^{-3}	15	9.6×10^{-3}	30
12	3.3×10^{-4}	2.5×10^{-3}	8	4.6×10^{-3}	14
13	1.6×10^{-4}	2.6×10^{-3}	16	5.0×10^{-3}	31
14	1.5×10^{-4}	3.6×10^{-3}	24	3.1×10^{-3}	21
15	2.1×10^{-4}	2.6×10^{-3}	12	3.8×10^{-3}	18
16	2.2×10^{-4}	8.9×10^{-4}	4	3.2×10^{-3}	15

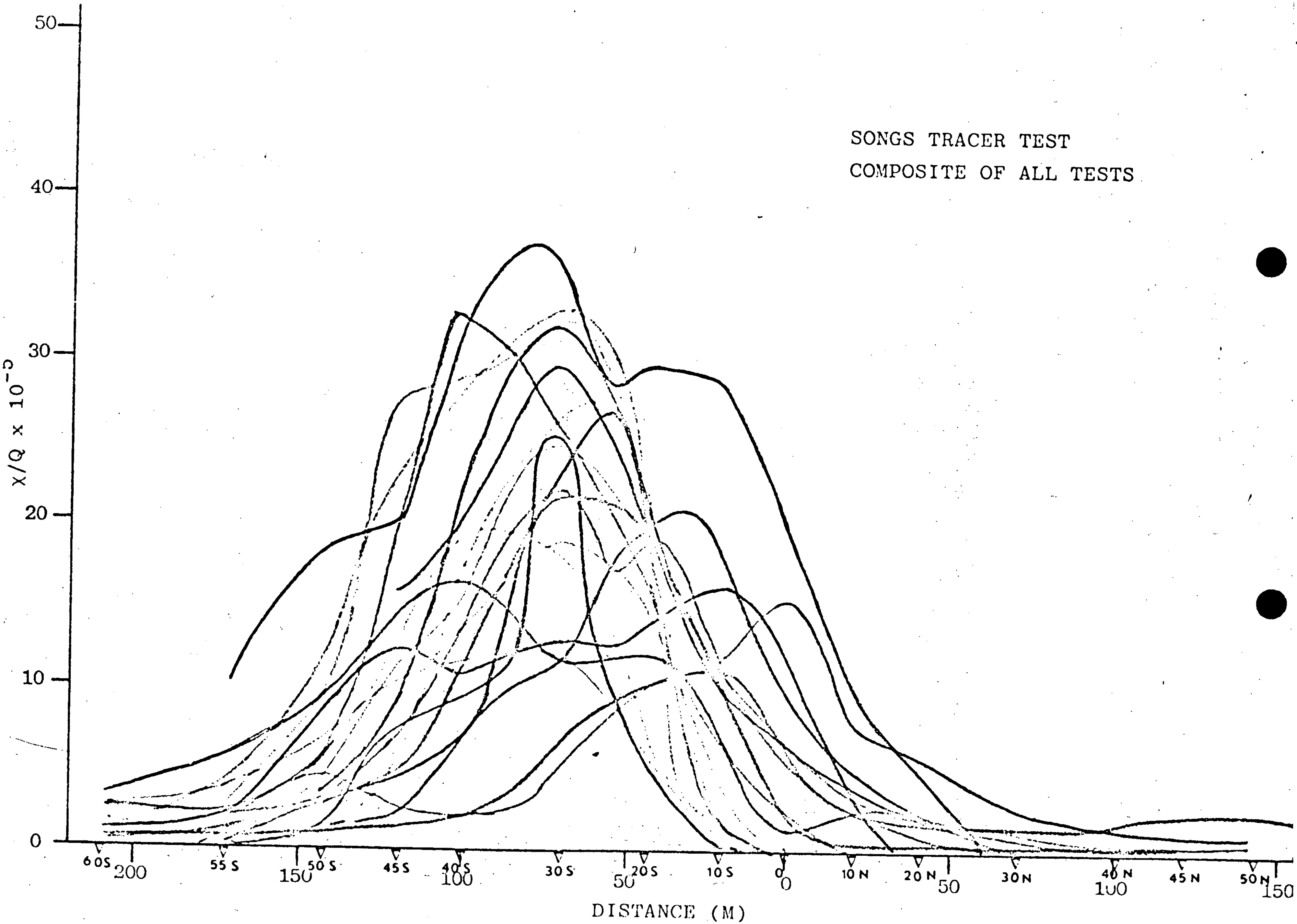


SONGS TRACER TESTS
TOP OF FUEL STORAGE BUILDING
χ/Q x 10⁻⁵
50
40
30
20
10
0
20
21
55 S
50 S
45 S
40 S
30 S
20 S
10 S
10 N
20 N
30 N
40 N
45 N
50 N
200
150
100
50
0
50
100
150
DISTANCE (M)

COMPARISON
OF CALCULATED AND MEASURED

X/Q
Elevated release - Fuel Storage Building

Test No.	Measured X/Q	Bluff Tower calculated	Bluff Tower red. factor	Beach Tower calculated	Beach Tower red. factor
20	2.7×10^{-4}	5.4×10^{-3}	20	1.3×10^{-2}	48
21	2.5×10^{-4}	1.0×10^{-2}	40	1.4×10^{-2}	56



SONGS TRACER TEST
COMPOSITE OF ALL TESTS
χ/Q x 10⁻⁵
50
40
30
20
10
0
60S
55S
50S
45S
40S
30S
20S
10S
0
10N
20N
30N
40N
45N
50N
200
150
100
50
0
50
100
150
DISTANCE (M)

PLATE 1

TIME BLOCKS

"72"	JANUARY 25, 1972 - JANUARY 24, 1973
"73"	JANUARY 25, 1973 - JANUARY 11, 1974
"74"	MARCH 31, 1974 - MAY 13, 1975
"75"	MAY 14, 1975 - JANUARY 24, 1976

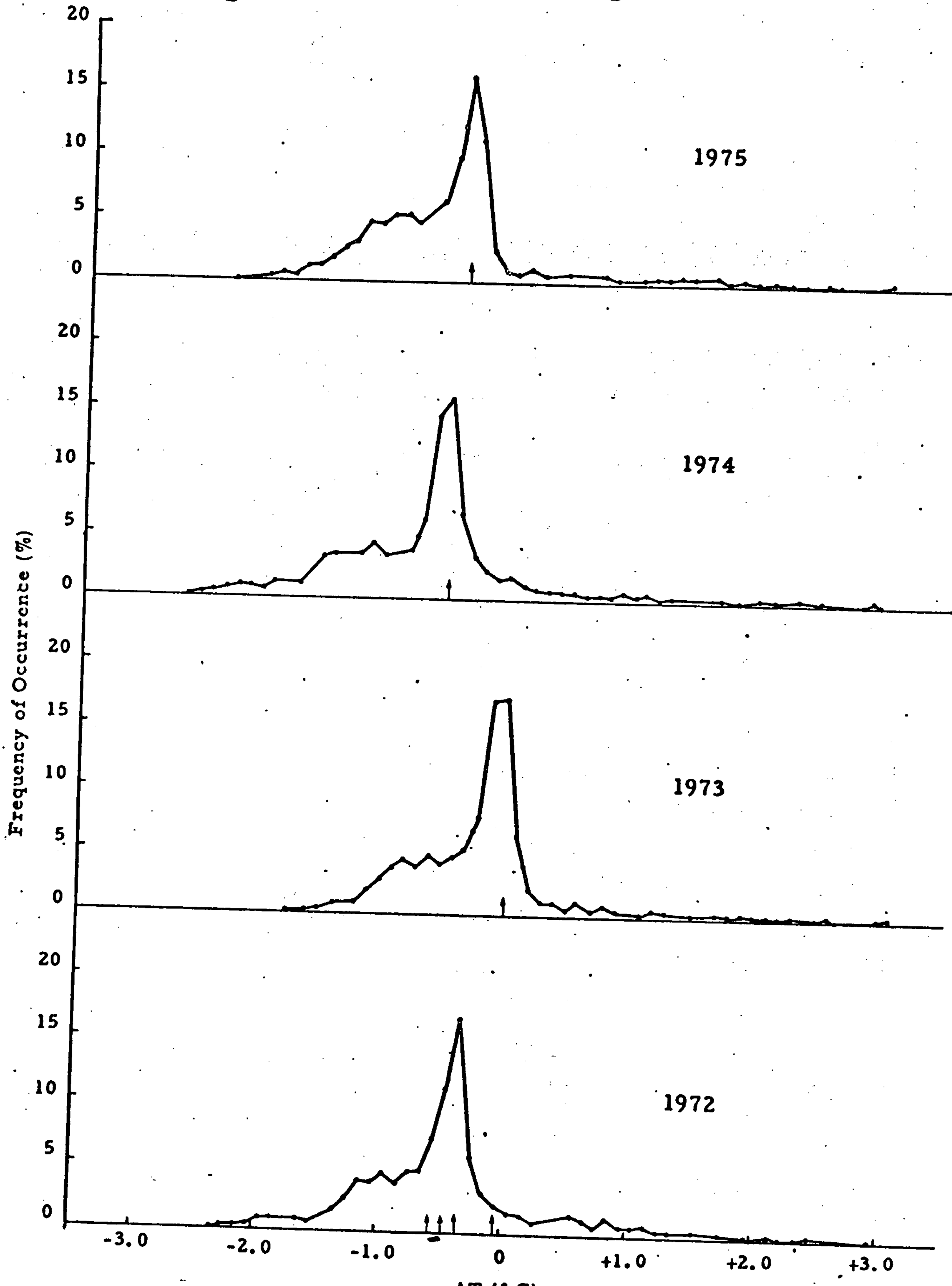
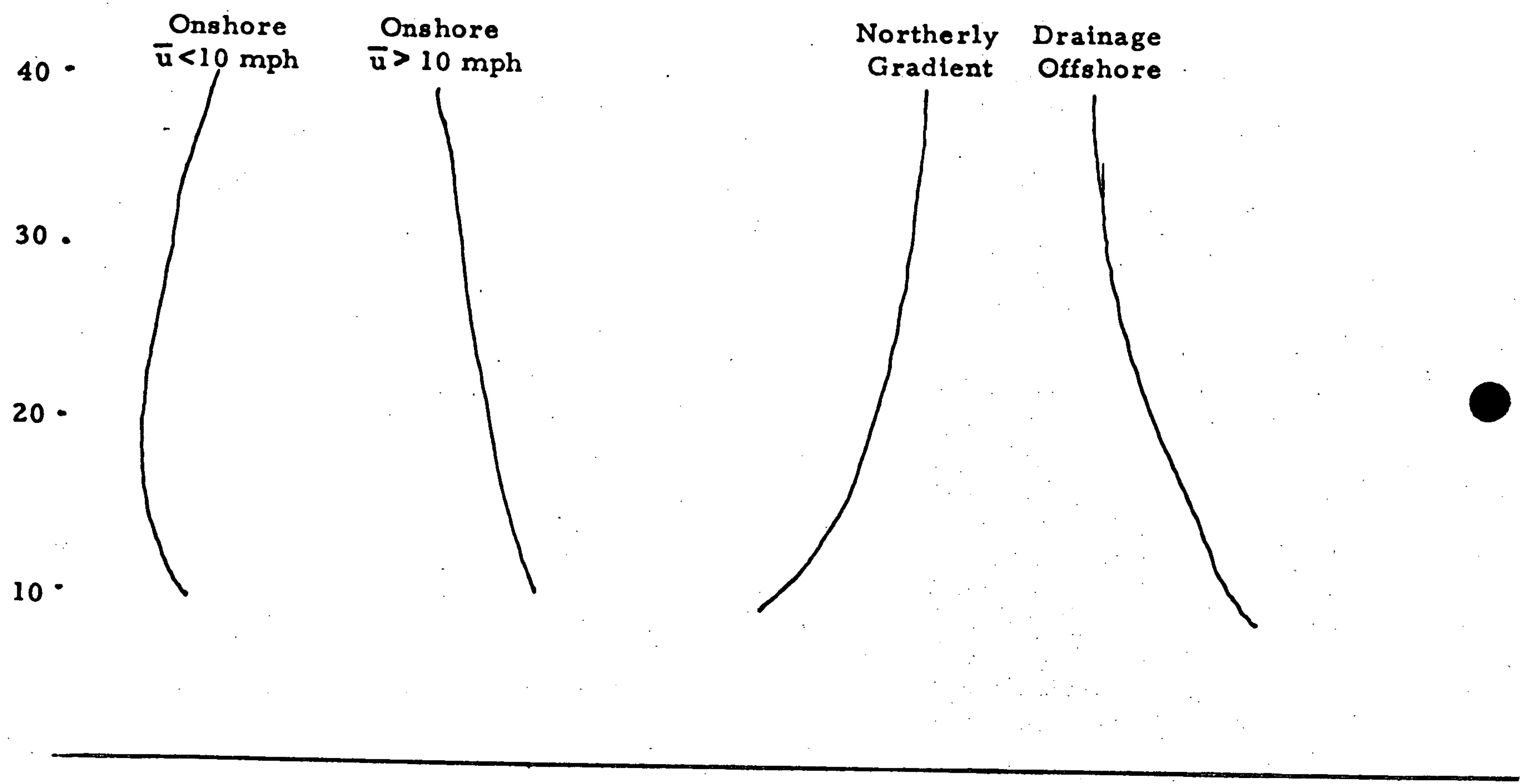
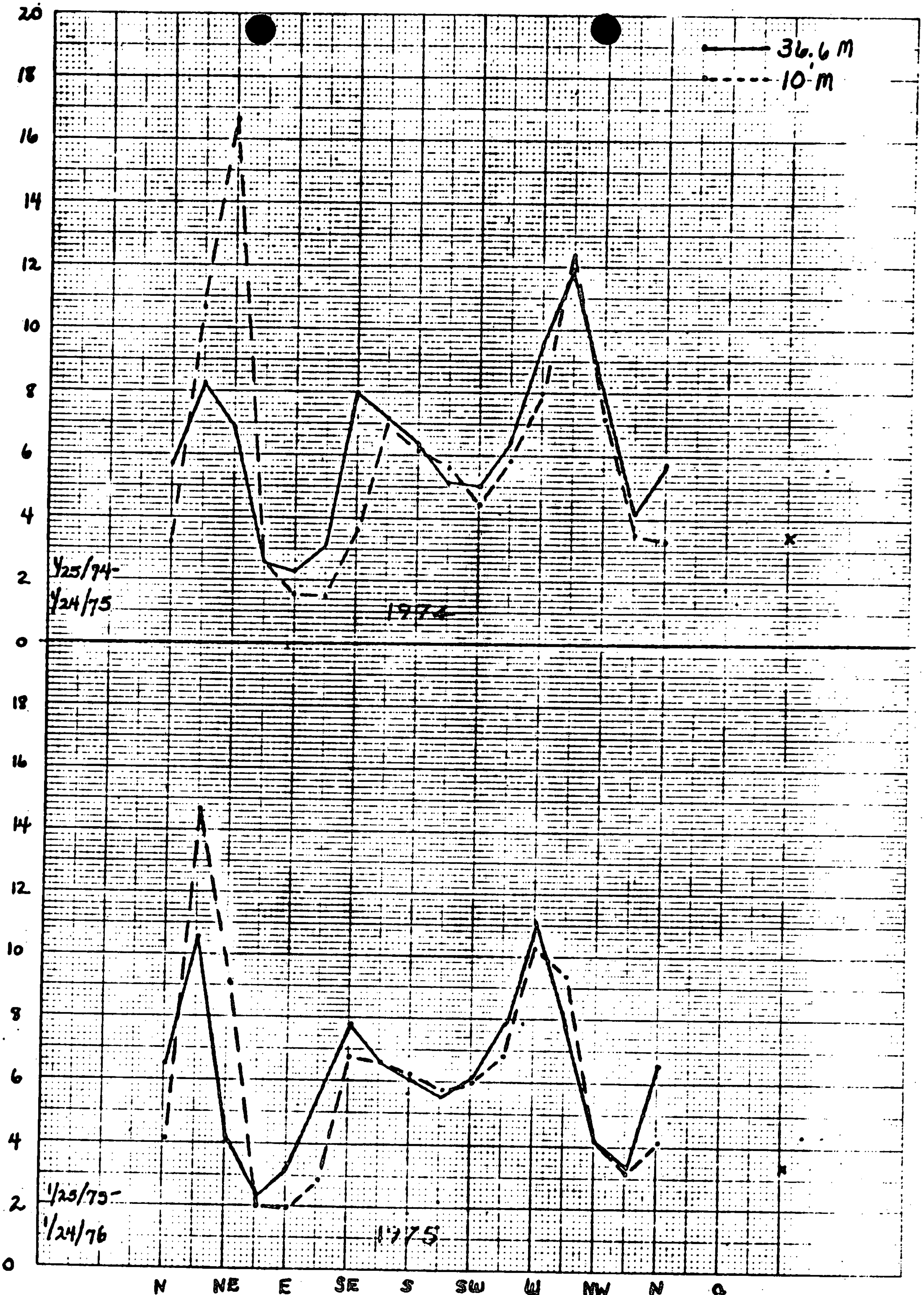


Fig. 2. ΔT DISTRIBUTIONS - SAN ONOFRE, JUNE 1 THROUGH AUGUST 31

WIND PROFILES





36.6 M
10 M
1/25/74-
1/24/75
1974
1/25/75-
1/24/76
1975
0
2
4
6
8
10
12
14
16
18
20
N
NE
E
SE
S
SW
W
NW
N
0

Fig. 3

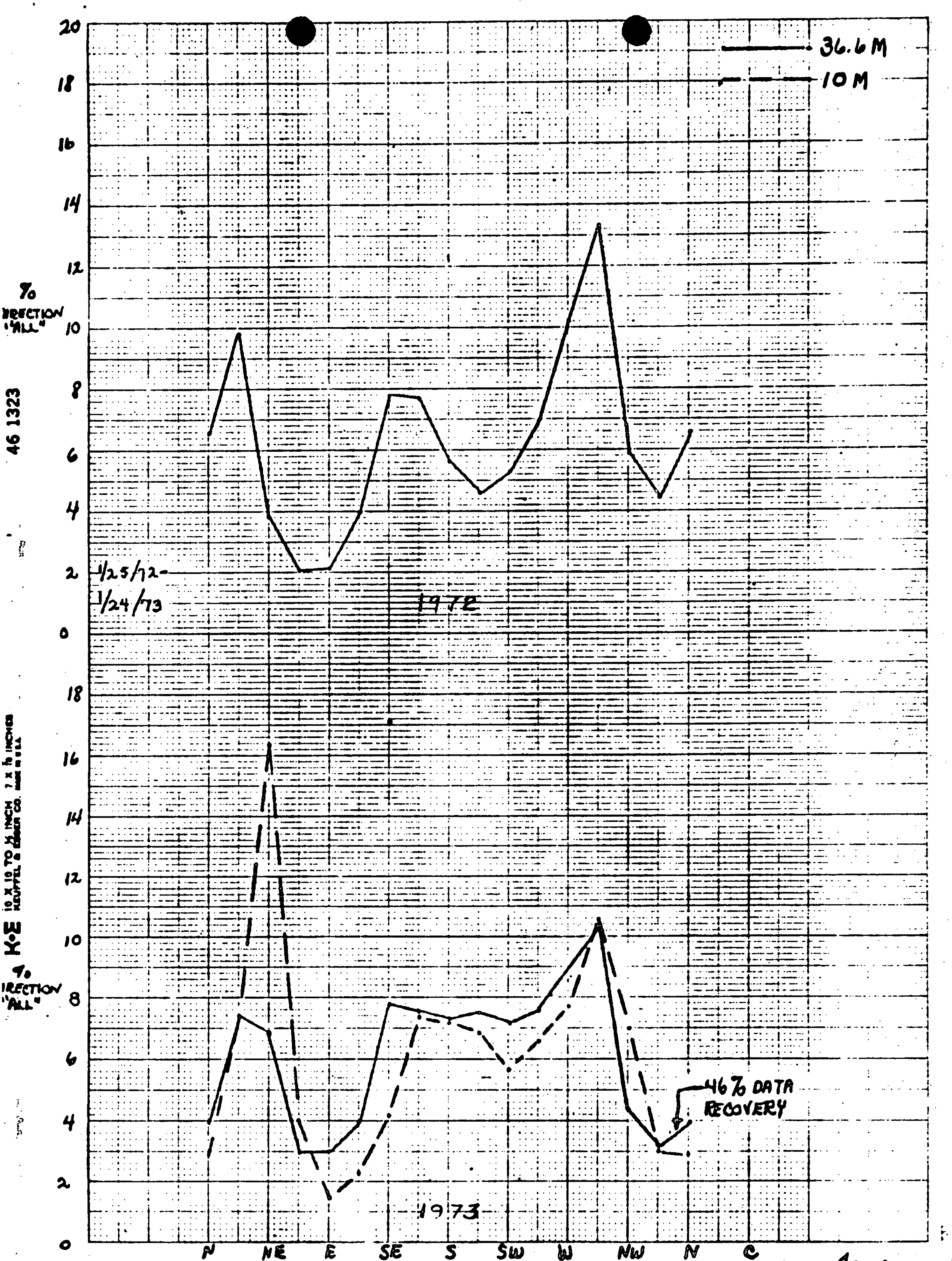

36.6 M
10 M
% DIRECTION "ALL"
1/25/72-
1/24/73
1972
1973
46% DATA
RECOVERY
N
NE
E
SE
S
SW
W
NW
N
C
0
2
4
6
8
10
12
14
16
18
20
46 1323
K+E 10 X 10 TO 1/2 INCH 7 X 10 INCHES
KEUFFEL & ESSER CO. MADE IN U.S.A.

Fig 2

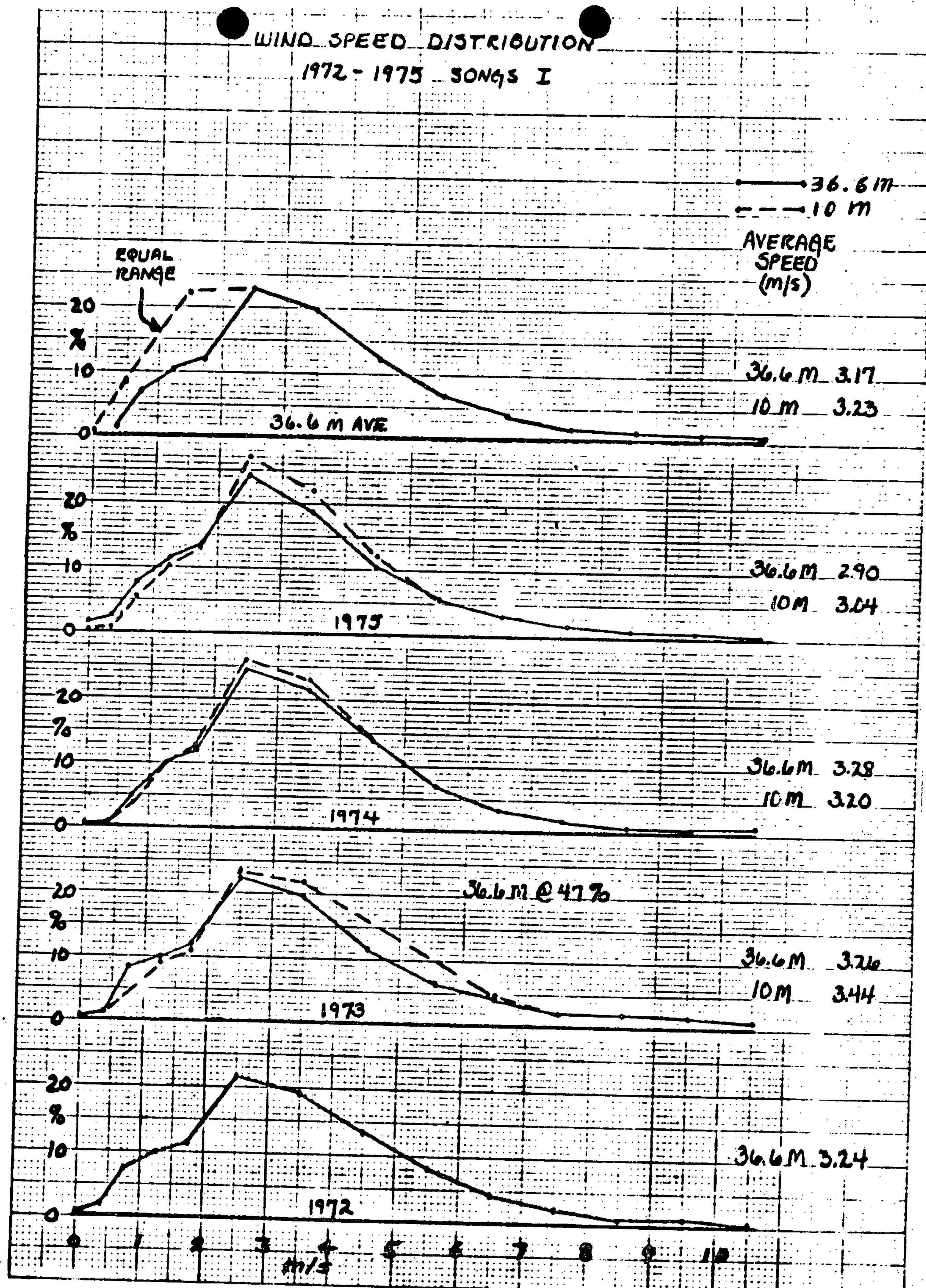
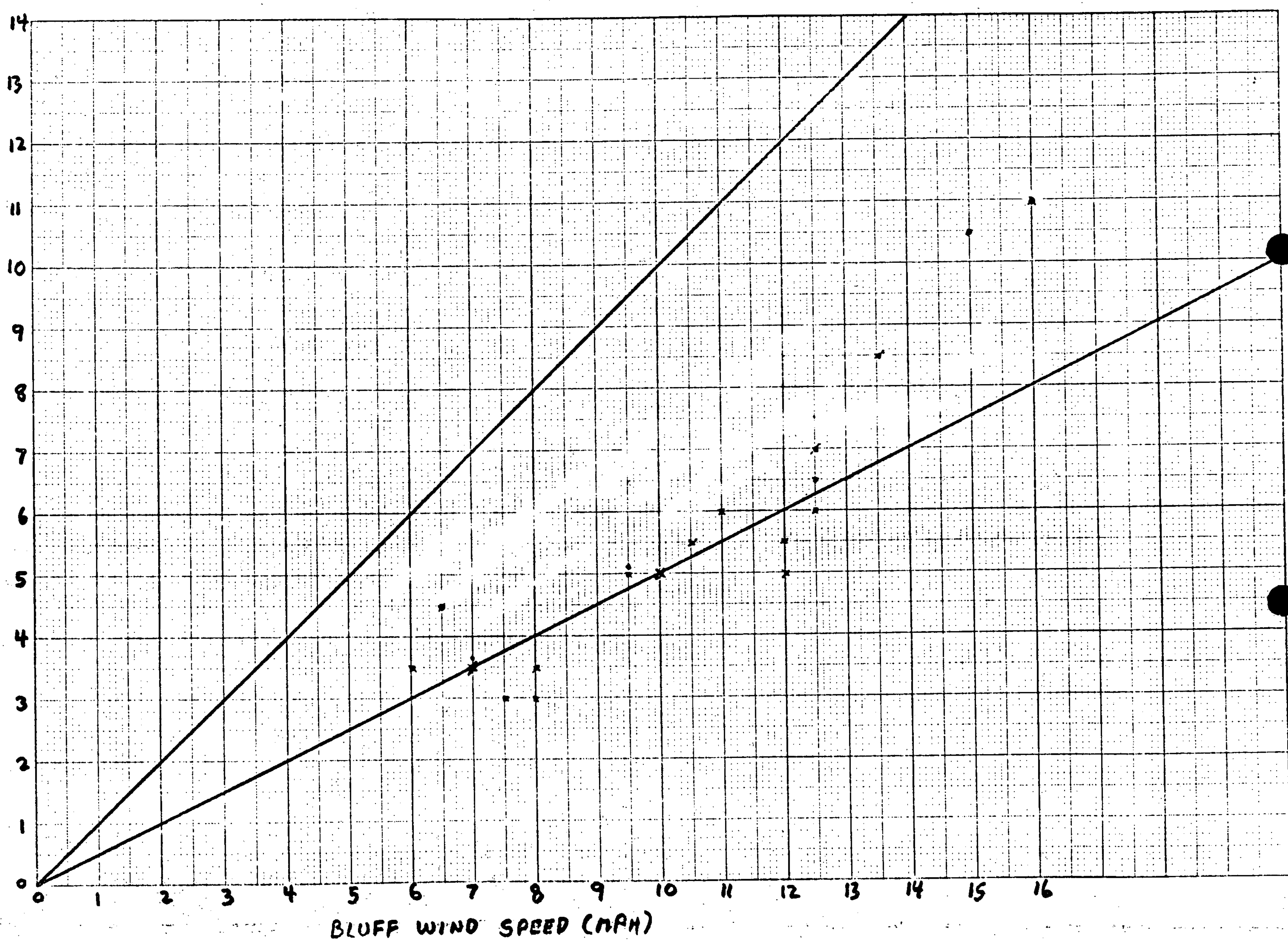


Fig 5

46 1323

K+E 10 X 10 TO 1/2 INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

		Bluff Tower Stability Category	
		F	G
Beach Tower Stability Category	F	0	1
	G	2	17



14
13
12
11
10
9
8
7
6
5
4
3
2
1
0
WIND SPEED (MPH)
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
BLUFF WIND SPEED (MPH)

SONGS

$$\frac{\chi}{Q} = \frac{1}{\bar{u}\,(\pi \sigma_y \sigma_z + cA)}$$

$$cA = 1790\ m^2$$

without the restriction that

$$\frac{\chi}{Q} \geq \frac{1}{3\,\bar{u}\,\sigma_y\,\sigma_z\,\pi}$$

$$P_{xj} = f_j \cdot P_{cj}$$

P_{xj} = probability of dilution factor x in sector j

f_j = relative frequency of occurrence of wind in sector j

P_{cj} = percentile of cumulative frequency distribution associated with dilution factor x in sector j

CENTERLINE CALCULATIONS. DELTA T ONLY.

[illegible]-DELTA T-STANDARD SCF - CIRCULAR BOUNDARY 1/25 - 1/24/74

CHI/Q VALUES FOR VARIOUS PROBABILITY POINTS FOR TIME PERIODS OF 1 HOURS.

DIRECTION FROM RELEASE POINT SSE .

DISTANCE FROM RELEASE POINT 150.0 METERS.

RELEASE HEIGHT 0.0 METERS.

RECEPTOR HEIGHT 0.0 METERS.

BUILDING WAKE FACTOR = 1060.5 METERS.

NUMBER OF OBSERVATIONS = 224 . TOTAL OBSERVATIONS = 7868 .

PROBABILITY IN PERCENT OF TOTAL OBSERVATIONS FOR THIS DIRECTION.

WORST CASE PROBABILITY = 4.4643E-01 CHI/Q VALUE = 1.9128E-02

PROBABILITY	CHI/Q
.4	1.9128E-02
.9	1.4015E-02
1.3	1.3155E-02
2.2	1.0511E-02
3.1	9.1973E-03
4.0	7.3578E-03
5.4	7.0075E-03
6.2	7.0075E-03
7.1	6.7509E-03
8.0	5.7383E-03
9.4	5.2618E-03
10.3	5.2556E-03
20.1	2.5056E-03
25.4	2.0134E-03
30.4	1.7389E-03
40.2	1.3192E-03
50.4	1.0318E-03
[illegible]	[illegible]

Docket cy

APR 2 1976

Docket Nos. 50-361
and ~~50-362~~

Olan D. Parr, Chief, Light Water Reactors Branch No. 3, DPM

FORTHCOMING MEETING WITH SOUTHERN CALIFORNIA EDISON - SAN ONOFRE 2 & 3

DATE & TIME:	Friday, April 16, 1976 - 9:00 a.m.
LOCATION:	Room P-114
PURPOSE:	Discuss SCE response to 3/31/76 request for additional information.
PARTICIPANTS:	SOUTHERN CALIFORNIA EDISON (Dwight Nunn, etc.)
	NRC - STAFF (J. Goll, E. Markee, C. Ferrell, L. Soffer, J. Sears, R. Froelich, E. Hawkins, R. Ross, S. Treby, H. Rood, etc.)

Original Signed by
Harry Rood

Harry Rood
Light Water Reactors
Branch No. 3
Division of Project Management

OFFICE ➤	DPM:LWR #3					
SURNAME ➤	HRood:mt					
DATE ➤	4/1/76					

FEB 18 1976

Docket Nos. 50-206
50-361
and 50-362

NRC PDR
Local PDR

ADDITIONAL INFORMATION CONCERNING METEOROLOGY - SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1

The letter dated December 31, 1975, from Mr. K. P. Baskin, Manager, Generation Engineering, Southern California Edison Company to Robert A. Purple, Chief, Operating Reactors Branch #1, U.S. Nuclear Regulatory Commission, provided additional meteorological information concerning the San Onofre site. The information included a data tape containing hourly meteorological tower observations between January 25, 1973, and January 24, 1975, based on delta temperature only. It was found impractical to place the data tape in the PDR; however, it will be made available to any interested party upon request.

Original signed by
R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors

bcc: J. Goll
E. Markee

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SURNAME ➤	ABurger:lb	RAPurple				
DATE ➤	2/18/76	2/ /76				

OCT 6 1975

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Docket Nos. 50-361
and 50-362

D. Skovholt, Assistant Director for Quality Assurance and Operations, RL

REQUEST FOR ASSISTANCE IN PREPARATION OF RESPONSE TO INQUIRY ABOUT SAN ONOFRE SITE EMERGENCY PLAN

The attached letter was forwarded to D. Kartalia, OELD, by Chairman A. Rosenthal of the San Onofre Units 2 and 3 Appeal Board. Chairman Rosenthal requested that we respond to Mrs. Hicks' inquiry.

In her letter Mrs. Hicks has requested information regarding the San Onofre Nuclear Generating Station emergency plan. The issue of emergency planning for the San Onofre facilities was the subject of detailed testimony at the San Onofre Unit 2 and 3 ASLB hearing.

We request the assistance of the Industrial Safety and Emergency Planning Branch in preparing a response to Mrs. Hicks' letter. We request that the draft response be provided by October 10, 1975. This matter has been discussed previously with R. Wayne Houston, Chief, Industrial Safety and Emergency Planning Branch.

Original Signed by
R. A. Birkel

for Olan D. Parr, Chief
Light Water Reactors
Project Branch 1-3
Division of Reactor Licensing

Attachment:
Letter dated October 18, 1975
from Nuclear Regulatory
Commission and ASLAB

cc: R. W. Houston
D. Kartalia
B. Grimes
J. Sears

Memo 4

OFFICE ▶	RL:LWR 1-3	RL:LWR 1-3				
SURNAME ▶	PDO'Reilly	ODParr for				
DATE ▶	10/6/75	10/6/75				

Nuclear Regulatory Commission and
Atomic Safety and Licensing Appeal Board
Washington D.C.

October 18, 1975

Docket Nos. 50-361 50-362

Dear Sirs:

This is a request for information necessary to the function of the responsibilities of the Evacuation and Emergency Planning Committee of the San Onofre State Park Citizens Advisory Commission.

As the Citizens Advisory Commission is appointed by the Director of the California State Parks and Recreation Department, our recommendations for evacuation and emergency planning are to be made to Director Herbert Rhoades, and the information we gather has thus-far been obtained from our local and area parks officials and from the public records. This data is inadequate to the extent that it is impossible for us to carry out our assignment. We ask your clarification of essentials.

First, I have made a detailed review of the document titled, "Evacuation Plan for Area Surrounding San Onofre Nuclear Generating Station" which is the beginning of evacuation planning for the area. We find no time parameters, and both the state parks department officials responsible for our local evacuation input to the "plan" say there are none in writing, that they have only scant verbal estimates from the utility officials.

Second, the verbal estimates on which the "plan" was based are tremendously at variance with estimates given in testimony before the AEC in contruction permit hearings, by both Edison Co's witness, evacuation expert Sheppard, and by AEC Lead Safety Engineer John Sears:

1. Mr. Ronald Hanshew, executive officer of our local state parks area, said the plan is predicated on the understanding from the utility officials that Interstate 5 north would be available for evacuation of the Parcel One (north beach) populations, that "it would not be necessary to evacuate San Clemente".

2. Sheppard's AEC hearing testimony was to the contrary: that Interstate 5 would be full of San Clemente evacuees, thus could not be used by the north beach area populations.

3. Similarly AEC's Sears estimates, based on typical wind velocity and assuming the winds constant, and also assuming that the accident which caused release of radioactivity from the plant, (ie: aircraft impact, earthquake, sabotage or enemy attack) had simultaneously knocked out the iodine removal spray systems, was that there would be 45 minutes in which to evacuate the children of Concordia School three miles from the plant, and that at 15 miles the people must be evacuated from a six mile width of the plume of radiation. His assessment allows 15 minutes for the first mile...not for the few hundreds of feet or yards in which beach goers and surfers must be evacuated.

4. Paul Muspratt, responsible for evacuation planning for the state parks in the southern California area, told us he has communicated to his superiors his assessment that "there is no way we can get the people off those beaches and out of the ocean and

out of the danger area in the 15 minutes they gave me for the first quadrant." He said, "I just don't have the manpower nor the equipment to perform that kind of evacuation."

The 15 minute time estimate was verbal, according to Mr. Muspratt. Were Sheppard and Sears in error? Or have the utilities proposed additional mechanical safeguards which have diminished the hazards to the extent that Sears' and Sheppards' estimates are no longer valid? If so, what changes have been judged to provide that additional time? Exactly what would be the minimum time available for evacuation within the exclusion area and within the low-population zone?

Are there other mechanical changes which the utilities could make which would allow more time? What about the people in the barrancas and on the beach in front of them, within a few hundred feet of the reactors, could they be given even the 15 minutes?

One of our committee members suggested that if the utilities provided a patrol boat beyond the outer surf break, it could pick up the surfers far out in the ocean. This might be a partial solution for the surfing beach area. Additional manpower and equipment, such as jeeps, might enable Mr. Muspratt and Mr. Hanshew to transport people from the pinnacles area, and elderly people or small children from the camping beach near the plant. We cannot intelligently make such suggestions until we know the time and distance constraints within which we must plan.
An evacuation plan is worthless, without accurate time parameters. We must have these.

Does our government require the utilities to take the responsibility for evacuating any person beyond their plant boundaries? If so, to what distance are they responsible? Mr. Hanshew said it sounded as though the extent of the utility company effort in the barranca, beach area and surf would be loud-speaker systems.

Our planning is dependent on what manpower and equipment assistance is immediately available. Please inform us thenumber of persons and how much time would be required for them to arrive in the barrancas beach.

In your December decision re San Onofre you affirmed doubts of the adequacy of roadways available for evacuation, based on the AEC hearing testimony. Specifically, what do you judge is needed? Can we expect that you will require the utilities to widen and lengthen roads as necessary for evacuation? We do not have funds for such projects. If the utility companies are not required to provide them, from what source do you anticipate they will come?

We will appreciate your assistance. I am anxious to have detail and specifics gathered in time for a meeting of my committee on October 2.

Sincerely,

Lyn Harris Hicks

Lyn Harris Hicks, Chairman, Evacuation and Emergency Planning Committee
San Onofre State Park Citizens Advisory Commission

Docket Nos. 50-206
50-361
50-362

AUG 25 1975

NOTE TO: NRC PDR
Local PDR

METEOROLOGY DATA FOR THE SAN ONOFRE SITE

The letter from Southern California Edison Company dated August 13, 1975, to Mr. Robert A. Purple, Chief, Operating Reactors Branch #1, Division of Reactor Licensing, submitted chronological listing of hourly meteorological tower observations between January 24, 1973 and January 25, 1975, and long and short term stability distribution from the National Weather Service Stations at Los Angelos and San Diego. It was found impractical to place this information in the PDRs; however, it will be made available to any interested party upon request.

/s/

Alfred Burger
Operating Reactors Branch #1
Division of Reactor Licensing

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ABurger
SMSheppard

MEMO

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SURNAME ➤	ABurger:dc					
DATE ➤	8/25/75					

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50-362
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file

Docket Nos. 50-361 and 50-362

MAR 27 1975

Richard C. DeYoung, Assistant Director
for Light Water Reactors, Group 1
Division of Reactor Licensing

TECHNICAL ASSISTANCE REQUEST - SAN ONOFRE UNITS 2 AND 3, REVIEW OF SOUTHERN CALIFORNIA EDISON'S PROPOSAL FOR "PROTOTYPAL TESTING OF SAN ONOFRE UNIT 2 CONTAINMENT STRUCTURE" (TAR-1462)

Title: Prototypal Testing of San Onofre 2 Containment Structure
Originating Organization: Southern California Edison Company
Description of Request: Technical Assistance Request Control form
Requested Completion Date: March 28, 1975
Branch and Project Manager Requesting Assistance: LWR Branch 1-3, P. D. O'Reilly

Review Status: Complete

In accordance with your request, the Structural Engineering Branch, Division of Technical Review has reviewed the subject proposal. Our review conclusions as well as the technical bases for our conclusions are provided in the enclosure.

R. R. Maccary, Assistant Director
for Engineering
Division of Technical Review

Enclosure: As Stated

cc w/o encl:
A. Giambusso

cc w/encl:
W. G. McDonald
S. H. Hanauer
F. Schroeder
O. D. Parr
L. C. Shao
P. D. O'Reilly
D. C. Jeng
A. Gluckmann

memo

OFFICE ➤	TR:SEB 7807	TR:SEB	TR:SEB	TR:AD:E		
SURNAME ➤	DCJeng:mb	AGluckmann	LCShao	RRMaccary		
DATE ➤	3/26/75	3/26/75	3/27/75	3/27/75		

MAR 27 1975

TECHNICAL EVALUATION OF SOUTHERN
CALIFORNIA EDISON CO.'S DOCUMENT TITLED
"PROTOTYPAL TESTING OF SAN ONOFRE
UNIT 2 CONTAINMENT STRUCTURE"

The Regulatory staff has reviewed the subject document for assessing the acceptability of the applicant's proposed exceptions to the requirements of Regulatory Guide 1.18. Our conclusions of the review are given in Part I of this evaluation, whereas, the technical basis for arriving at the conclusions are discussed in Part II below:

Part I Conclusions

1. The applicant's proposed structural acceptance test program for prototypical San Onofre Unit 2 containment structure at the springline and the dome regions is acceptable.
2. The applicant's proposed ommission of the strain measurements in the vicinity of the anchor for the U-shaped tendon (dome plus vertical tendons) is acceptable.
3. All the other provisions of the Regulatory Guide 1.18 should be complied with by the applicant including the special provisions for prototypal testing.

Part II Discussions of Technical Bases

1. Number of Buttresses

   The discussion and conclusion given on pages 7 thru 10 of the document related to the need for obtaining strain measurements for a three buttress containment are judged as not necessarily pertinent and correct in its entirety, therefore, the proposed exception to item 2.a, Appendix A of the Regulatory Guide 1.18 is not acceptable.

   Specifically, the following reasons are cited for the staff's conclusion:

a. The purpose of the test is not only to verify the expected close agreement between the design analysis results and the actual measured response of the containment structure, but also to establish the soundness of the "as built" structure. The Regulatory Guide 1.18 has been prepared with this thought in mind and covers as many locations of the structure as practicable, where the construction weaknesses may show up during and after the construction.

b. The staff feels that the six buttresses containment of Turkey Point and Palisades type would result in a more axisymmetric structure (therefore, more amenable to the use of axisymmetric computer code in structural analysis) than the three buttresses type containment, insofar as the vertical containment shell stiffness is concerned. Additionally, the increased length of hoop tendons, covering 240 degrees of circumference, may cause frictional losses that are quite different from those of the six buttresses containment.

c. Referring to page 8, second paragraph, which states that "The predicted analytical response of ... which ignores the buttress effect altogether ", it is noted that the intent of this statement requires further clarification. Discuss your intent for comparing two sets of "predicted" results. Also clarify the criterion for judging that two predicted results are "consistent". Ordinarily, one would compare the predicted results with those actually observed to justify the validity of the prediction. The applicant should clarify this portion of the discussion.

d. In mid portion of page 8, it is stated that "These results show little variation in measured response between the typical wall cross-section and a buttress for the region of membrane stress." This statement is not necessarily correct, e.g.,

if one refers to Figures 6-28 and 6-32, one sees significant differences in measured outside hoop strains for all cases shown in the figures at mid height point of the containment shell. (for the case under prestress alone -150 micron (typical) versus - 260 (buttress), for the 1.15 design pressure cases; +110 (typical) versus +20 (buttress); and for the prestress plus 1.15 design pressure; virtually zero strain (typical) versus large compressive strain -220± (buttress)). At the discontinuity regions both the measured and predicted responses are found to differ significantly as can be seen from the Figures.

The conclusion drawn by the last paragraph on page 8 is not necessarily correct.

e. With respect to last portion of page 9, strain data obtained at the Palisades plant (6 buttresses type) is quoted in connection with the discussion of Attachment (3) data pertaining to a 3 buttresses containment. This mingling of incompatible data tends to introduce additional confusing factors to the issue being considered and therefore is judged inappropriate. Also the third figure of Figure 6-49 shows comparison of radial displacements computed from measured strains to the "Average" of wall and buttress sections' measured radial displacements. It should be noted that such a comparison, in general, has no major significance. One should compare the displacements obtained from the measured strains to the displacement data obtained from pertinent cases rather than the "averaged" data, in order to be meaningful in drawing conclusions.

2. Size of Tendons

The proposed exception to item C.5.C requirement of the Regulatory Guide 1.18, that the vertical, horizontal and shear strains in the concrete should be measured under one prestressing anchor of a vertical tendon, is acceptable to the staff. The basis for accepting this exception is that since all the anchor points of vertical tendons are located in the prestressing gallery under the base mat and the stresses in the anchor region cannot be a problem due to presence of massive concrete resisting the anchor loads. Also, Bechtel topical reports BC-TOP-7 and 8 provide additional basis for the staff acceptance. These two reports address the subjects of full scale buttress test for prestressed containment structure and tendon end anchor reinforcement post, respectively, and have previously been reviewed and accepted by the staff.

3. Dome Configuration and Associated Modifications

As indicated in the item 1 discussion given above, the proposed exception is not acceptable.

4. Strain Measurements at Top of Basement on One Meridian

On page 14 mid-portion, it is stated that "...1) that the analytical techniques utilized ... in the base mat region and 2) that the SIT responses can be correlated with analytical predictions." This statement is not necessarily correct. There seem to be considerable variations between the measured and the predicted responses at the base mat region in some of the figures presented (e.g., figures 6-30, 6-33 and 6-34). For this prototype structure, the strain measurements at top of base mat will be required. The purpose of this measurement is to permit a check in design and construction at this interface point. In

the present case, this requirement is extremely important since no detailed knowledge exists on behavior of long U-shaped tendons in containment structures.

5. Strain Measurements at Largest Opening

Strains should be measured at the largest opening. This provision has been included in the Regulatory Guide 1.18 to check on the behavior of tendons draped around the opening. In the present case this provision is of the utmost importance for the same reason as stated in item 4 above. The applicant should comply with this requirement.

6. Strain Measurements at Springline Level of the Dome

The applicant's strain measurement procedure discussed in item V.C (page 15) is acceptable to the staff.

7. Strain Measurements at End Anchorage Region of a Vertical Prestressing Tendon

The applicant's proposed exception in item V.D is acceptable.

8. Extent of Instrumentation

Strain measurements should be made at three positions within the wall per item C.5.c of the Guide. The aim of this provision is to detect any non-linear strain or unusual strain distribution that may be caused by unexpected construction weaknesses. The applicant should comply with this requirement.

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Docket Nos. 50-361
and 50-362

FEB 28 1975

F. Schroeder, Acting Director, Division of Technical Review, NRR
THRU: R. C. DeYoung, Assistant Director for Light Water Reactors Group 1

TECHNICAL ASSISTANCE REQUEST FOR EVALUATION OF PROPOSED CONFORMANCE WITH REGULATORY GUIDE 1.18 - SAN ONOFRE UNITS 2 AND 3

On January 30, 1975, representatives of Southern California Edison Company (SCE) and Bechtel met with the NRC staff in Bethesda. SCE had requested the meeting to discuss the degree of conformance with the recommendations of Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments".

SCE stated that the San Onofre Units 2 and 3 PSAR had committed to comply with Regulatory Guide 1.18. SCE explained that in the final design of the containment dome, a hemispherical dome was used instead of the original ellipsoidal configuration. Due to construction and schedule delays incurred by other applicants, the San Onofre Units 2 and 3 containment design has become a prototype for purposes of complying with Regulatory Guide 1.18. SCE stated that they were proposing a technique for meeting the recommendations of Regulatory Guide 1.18 which would provide demonstration of acceptability under Regulatory Guide 1.18 only for features unique to the San Onofre Units 2 and 3 containment design. We agreed to evaluate the proposed program for compliance with Regulatory Guide 1.18 in a timely manner upon submittal of the detailed information and to inform SCE of the results.

With their letter dated February 21, 1975, SCE submitted details of their proposed program for compliance with Regulatory Guide 1.18. We request that the Structural Engineering Branch review the document, "Prototypical Testing of the San Onofre Unit 2 Containment Structure" and provide us with an evaluation of its acceptability by March 28, 1975.

Original Signed by
Olan Parr
Olan D. Parr, Chief
Light Water Reactors
Project Branch 1-3
Division of Reactor Licensing

Enclosure:
As stated

cc: L. Shao

Mems

OFFICE	RL:LWR 1-3	RL:LWR 1-3	RL:AD/LWR 1
SURNAME	PDO'Reilly:cls	ODParr for	RCDeYoung
DATE	2/28/75	2/28/75	2/28/75

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Docket Nos. 50-361
and 50-362

FEB 28 1975

F. Schroeder, Acting Director, Division of Technical Review, NRR
THRU: R. C. DeYoung, Assistant Director for Light Water Reactors Group 1

TECHNICAL ASSISTANCE REQUEST FOR EVALUATION OF PROPOSED CONFORMANCE WITH REGULATORY GUIDE 1.18 - SAN ONOFRE UNITS 2 AND 3

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Original Signed by
Olan Parr

Olan D. Parr, Chief
Light Water Reactors
Project Branch 1-3
Division of Reactor Licensing

Enclosure:
As stated

cc: L. Shao

memo

OFFICE	RL:LWR 1-3	RL:LWR 1-3	RL:AD/LWR 1			
SURNAME	PDO'Reilly:cls	ODParr	RCDeYoung			
DATE	2/ /75	2/ /75	2/ /75			

Docket Nos. 50-206
50-361
50-362

FEB 25 1975

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K. R. Goller, Assistant Director for Operating Reactors, RL

EMERGENCY PLANNING CONSIDERATIONS AT SAN ONOFRE

During the course of review of the application for construction permits for San Onofre Units 2 and 3, information was developed concerning planned development of park and beach areas near the site. The plans call for development of the area as a State Beach, by the California Department of Parks and Recreation. The plans also call for development of camp sites and other facilities for the use of surfers and campers.

It seemed to us appropriate to consider this information with respect to the current operation of San Onofre Unit 1. The attached memorandum of John Sears dated February 25, 1975, evaluates the present beach usage vis-a-vis the emergency planning considerations.

We have concluded that at the present time there is no undue risk associated with the continued operation of San Onofre Unit 1. We plan, however, since some of the information that Mr. Sears relied on was obtained orally, to require written confirmation of this information from the licensee so that the public record is complete. Resolution of the question of future usage of park and beach areas near the site also will be necessary.

Original Signed by.
Donald J. Skovholt

Donald J. Skovholt
Assistant Director for Quality
Assurance and Operations
Division of Reactor Licensing

Enclosure:
As stated

cc: E. G. Case
A. Giambusso
R. W. Houston
J. R. Sears
D. E. Kartalia

Memo
(4)

ext 7492

OFFICE	RL:ADQAO					
SURNAME	DJSkovholt:jll					
DATE	2/25/75					

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

Docket No. 50-206

FEB 25 1975

Note to: Donald J. Skovholt, Assistant Director for Quality Assurance and Operations, RL
THRU: R. Wayne Houston, Chief, Industrial Security and Emergency Planning Branch, RL

RE: REGULATORY STAFF POSITION ON ACCEPTABILITY OF EMERGENCY PLANNING FOR SAN ONOFRE 1

San Onofre 1 station is equipped with a siren mounted on a tower southeast of the plant. The siren goes off automatically when the safety injection system actuates; the siren can also be switched on manually by the reactor operator. Mr. H. Ottoson, the Plant Superintendent, states that the siren can be heard up and down the beach 800 meters from the plant.

On top of the sea wall are located 4 loudspeakers. Mr. Ottoson states that whenever the siren is tested, a prior announcement is made on the public address system, and that this announcement can be heard approximately 100 meters up and down the beach.

Mr. Ottoson also states that, during peak use of the beach in the summer at the present time, there are no campers, beach-sitters or fishermen along the beach within the 800 meter exclusion radius of the plant. He estimates that, at any time of peak use, the maximum number of beach strollers within the exclusion radius would be 12 people. Excavation for Units 2 & 3 has been started and mounds of sand have been piled on the beach in front of the location of Units 2 & 3, and this has discouraged strollers from in front of the plant. He estimates that the access path, with fence and gates, along the beach around the Units 2 & 3 construction site, which must be provided in compliance with California Coastal Zone Conservation Commission Permit, will not be installed until the summer of 1975.

The only vehicles that Mr. Ottoson has seen on the beach in the vicinity of the plant have been Marine or California Recreation Department vehicles, because regular tired vehicles would probably get stuck in the sand. The hard packed dirt road along the beach which approaches the plant from the north peters out about 800 meters north of the plant.

The California State Department of Parks and Recreation has now a toll booth at the northern entrance to the park area. In an emergency, the attendant at the toll booth would be alerted to warn campers out of the area.

At the public hearing for San Onofre 2 & 3, the applicant's consultant traffic engineer testified that the exclusion area could be evacuated out to the exclusion radius, by walking, in 20 minutes. An ISEP staff member has confirmed the conservativeness of this estimate by walking the beach, from in front of San Onofre 1, a half mile north in 15 minutes. From approximately 200 meters north of the plant extending northward, the cliffs, which are approximately 80 feet high, afford shielding, and when the beach road picks up, it is directly at the base of the cliffs.

The areas to be developed for camping are sufficiently distant from Unit 1 so that two hour doses in excess of the Part 100 values would not be received in the event of a DBA occurring. However, in the beach area adjacent to the station, doses greater than the Part 100 values could be received if protective measures were not taken. The Emergency Plan provides that if an accident occurs, an announcement will be made over the sea wall speaker system to instruct personnel on the beach to evacuate the area. Since the high dose rate is associated with a narrow plume width, a person would significantly reduce his exposure by just moving down the beach a few hundred yards; this would take only a few minutes. These factors, taken in conjunction with the sparse usage of the beach and the prevailing onshore winds in daytime, make the risk associated with the beach very small.

The Regulatory staff has investigated the adequacy of existing evacuation routes both within the LPZ and beyond to a distance of 5 miles from the plant, and has concluded that there are sufficient routes of adequate capacity to allow orderly evacuation if required following an incident at San Onofre 1. We have also reviewed Amendment 36 to the FSAR for San Onofre 1, dated January 20, 1975, which includes the revised Emergency Plan for San Onofre 1. The Plan does not conform in all details to our current acceptance criteria and we are requesting the applicant to upgrade his Plan in these respects. The deficiencies in Amendment 36, however, do not refer to the capability of evacuating the plant's environs, and consequently do not lessen our conviction that adequate measures can and will be taken to protect the health and safety of the public in the unlikely event of an incident at San Onofre 1.

John R. Sears

John R. Sears
Industrial Security and
Emergency Planning Branch
Division of Reactor Licensing

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

FEB. 12 1975

DOCKET NOS: 50-361 AND 50-362

APPLICANT: SOUTHERN CALIFORNIA EDISON COMPANY (SCE)

FACILITY: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

SUMMARY OF JANUARY 30, 1975 MEETING WITH SCE REGARDING DESIGN BASIS CRITERIA FOR PIPE WHIP AT SAN ONOFRE 2/3

On January 30, 1975, representatives of SCE and Bechtel met with the Regulatory staff in Bethesda. SCE had requested the meeting in order to discuss the design basis criteria for pipe whip at San Onofre 2/3. Enclosure No. 1 contains a list of attendees.

By means of Mr. A. Giambusso's letter of December 18, 1972, the staff had requested SCE to evaluate breaks in high energy piping systems outside of containment at San Onofre 2/3, using criteria acceptable to the staff. These criteria were described in an enclosure to the Giambusso letter. In their responses of December 28, 1972 and February 23, 1973, SCE had committed to supply the information requested by the Giambusso letter in two parts. The results of an evaluation of breaks in the main steam and main feedwater systems piping outside containment were to be submitted by August 1974 and the corresponding information for breaks in the rest of the high energy systems piping was scheduled for submittal in December 1976 (with the FSAR). In Amendment No. 21 to the PSAR, submitted September 3, 1974, SCE provided the requested evaluation of breaks in the main steam and main feedwater systems piping outside containment. This amendment is currently under staff review.

SCE began their presentation by stating that they were proposing a consistent set of design basis criteria for pipe whip inside and outside of containment. Bechtel summarized the purpose and background for the pipe whip criteria. In addition to the Giambusso letter of December 18, 1972, the letters from Mr. J. O'Leary to certain applicants in April, 1973, Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," and the current Mechanical Engineering Branch position have presented methods of addressing pipe whip that are acceptable to the staff. Bechtel pointed out differences between Regulatory Guide 1.46 and the criteria in the Giambusso letter. Bechtel's interpretation and proposed resolution

memo (4)

of all the different criteria were outlined. Bechtel then summarized the pipe whip criteria contained in the Giambusso letter. The content of Amendment 21 was also summarized. Bechtel informed the staff that new data are available and have been evaluated. Based on this evaluation, Bechtel has revised the design basis criteria for pipe whip. These criteria, when submitted on the San Onofre 2/3 docket, would supercede the information on design criteria for pipe whip submitted in previous Amendments (10, 13, and 21). This decision was influenced by Bechtel's interpretation of information contained in the staff's position on pipe whip design criteria dated September 23, 1974.

Bechtel then described the proposed design basis pipe whip criteria (see Enclosure No. 2). The definition of high energy piping to be used for San Onofre 2/3 was discussed. The postulated locations for pipe breaks were outlined. Bechtel described the types of breaks that are postulated. The differences between the criteria presented in Amendment 21 in response to the Giambusso letter and the proposed criteria were explained by Bechtel (see Enclosure No. 3). Exceptions proposed to Regulatory Guide 1.46 in the new criteria were discussed (Enclosure No. 4). Bechtel proposes using a design condition that is higher than the upset condition and will yield a conservative solution.

The staff commented that the proposed criteria represent a reasonable approach. The staff agreed to review the proposed criteria and the exceptions to Regulatory Guide 1.46 and inform SCE about the acceptability of the criteria. The question of whether the new criteria should be submitted as an Amendment to the PSAR or submitted with the FSAR was discussed. The staff agreed to inform SCE which method of submittal was preferable.

Patrick D. O'Reilly

Patrick D. O'Reilly
Light Water Reactors
Project Branch 1-3
Division of Reactor Licensing

Enclosures:
1. Attendance List
2. Proposed Criteria for Postulated Breaks in High Energy Piping for San Onofre 2/3
3. Additions/Exceptions/Clarifications to Mr. Giambusso's letter of December 1972
4. Additions/Exceptions/Clarifications to Regulatory Guide 1.46

FEB. 12 1975

cc w/encl:
Mr. Jack B. Moore
Rollin E. Woodbury
Chickering & Gregory
Mr. Larry E. Moss
Mr. David Sakai
Frederick P. Sutherland, Esq.
Mr. Kenneth E. Carr
Alan R. Watts, Esq.
Lawrence Q. Garcia, Esq.
George Spiegel, Esq.

ENCLOSURE NO. 1

MEETING WITH

SOUTHERN CALIFORNIA EDISON

JANUARY 30, 1975

SOUTHERN CALIFORNIA EDISON

D. Hunn
J. Ploecker
W. C. Moody
D. F. Martin

BECHTEL

S. Sedin
T. Stick
L. Shipley
F. V. Naugle
C. S. Welty

COMBUSTION ENGINEERING

R. W. DeVanc
T. E. Natan

NRC - STAFF

P. O'Reilly
I. Villalva
K. D. Desai
J. M. Kovacs
R. J. Bosnak
H. L. Brammer
G. C. Millman

Enclosure No. 2
Proposed Criteria for Postulated
Breaks in High Energy Piping
for San Onofre Units 2 & 3

1. Definition of high energy piping:

Piping that, during normal plant conditions is either in operation or maintained pressurized under conditions where either (or both) of the following conditions are met:

a) maximum operating temperature exceeds 200°F, or

b) maximum operating pressure exceeds 275 psig;

Piping is not considered high energy if the piping run or branch run is high energy for less than 2% of the time it operates below both parameters defined above; i.e., 200°F and 275 psig. (As an example, piping in the Residual Heat Removal system would not be treated as high energy piping, while piping in the Auxiliary Feedwater System qualifies as high energy piping.)

2. Break locations:

If a review of the pipe layout and plant arrangement drawings by the plant designer shows that the effects of postulated piping breaks, on a reasonable basis, are isolated, physically remote, or restrained by plant design features from essential systems or components, then no piping breaks will be analyzed. Where required, postulated pipe breaks will be selected as described below and analyzed to demonstrate the capability to place the plant in a safe shutdown condition.

a) ASME Section III, Code Class 1 piping:

For ASME Section III, Code Class 1 piping, breaks will be postulated to occur at the following locations (piping weld joints to fitting, valve or welded attachment) in each piping run or branch run:

1) The terminal ends

2) At all intermediate locations where the primary plus secondary stress intensitities derived on an elastically calculated basis under the loadings associated with the operational plant conditions and an operating basis earthquake exceed 2.4 Sm.

1) The terminal ends

2) At all intermediate locations between terminal ends where the primary plus secondary stresses derived on an elastically calculated basis under the loadings associated with the operational plant conditions and an operating basis earthquake exceed 0.8 $(1.2\ S_h + S_A)$.

3) Where the stresses calculated for a particular piping run between terminal ends are at all points below the limits stated above, such that all intermediate pipe break locations would be considered unlikely, then the two intermediate locations exhibiting the highest stresses will be the locations where ruptures are postulated to occur. Additionally, if there is only one intermediate point where the stresses calculated exceed the value of 0.8 $(1.2\ S_h + S_A)$, one additional intermediate break will be postulated to occur at the location of the next highest stress intensity.

If stresses differ by less than 10%, two locations will be selected separated by a change of direction of the pipe run.

c) <u>Non Nuclear piping:</u>

Breaks in non nuclear piping will be postulated at the following locations in each run or branch run:

1) Terminal ends of the run.

2) At intermediate fittings (e.g. elbows, tees & reducers), welded attachment and valve.

3) At all intermediate locations where the cumulative usage factor, U, derived from the piping fatigue analysis under the loadings described above exceeds 0.1.

4) Where the stresses calculated for a particular run of piping between terminal ends are everywhere less than the stress limits stated above, such that all intermediate pipe break locations would be considered unlikely, then two intermediate locations of highest stress or usage factor will be chosen as the most likely break locations for piping runs longer than 10 diameters total length, and one intermediate location of highest stress or usage factor will be chosen as the most likely break location for piping runs shorter than 10 pipe diameters. Additionally, for piping runs longer than 10 pipe diameters, if there is only one intermediate point where stresses or usage factor exceeds the specified values, one additional break will be postulated to occur at the location of the next highest stress intensity. If stresses differ by less than 10%, two locations will be selected separated by a change of direction of the pipe run.

b) <u>ASME Section III, Code Class 2 & 3 piping:</u>

For ASME Section III, Code Class 2 and 3 piping, and for main steam and feedwater piping from the containment isolation valve and to the seismic anchor point, breaks will be postulated to occur at the following locations (piping weld joints to fitting, valve or welded attachment) in each piping run or branch run:

d) Fluid system piping between containment isolation valves:

Pipe breaks need not be postulated in piping between containment isolation valves (up to and including the pipe whip restraints which define the terminal ends for the run) when all of the following design requirements are met:

(1) For ASME, Section III, Class 1 piping

There are no Code Class 1 piping penetrations in San Onofre 2 and 3.

(2) For ASME, Section III, Class 2 piping

(a) The maximum stress ranges as calculated by Eq. (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering operational plant conditions (i.e., sustained loads, occasional loads, and thermal expansion) and an OBE event do not exceed 0.8 (1.2 S_h + S_A).

(b) The maximum stress, as calculated by Eq. (9) in Paragraph NC-3652 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping should not exceed $1.8S_h$.

(c) Welded attachments, for pipe supports or other purposes, to these portions of piping will be avoided except where detailed stress analyses, or tests, are performed to demonstrate compliance with the limits of 0.8 (1.2 S_h + S_A) or 1.8 S_h as applicable.

(d) The number of circumferential and longitudinal piping welds and branch connections will be minimized. Where guard pipes are used, the enclosed portion of fluid system piping will be seamless construction unless specific access provisions are made to permit inservice volumetric examination of the longitudinal welds.

(e) The length of these portions of piping will be reduced to the minimum length practical.

(f) The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) will not require welding directly to the outer surface of the piping (e.g., flued integrally forged pipe fittings may be used) except where such welds are 100 percent volumetrically examinable in service and a detailed stress analysis is performed to demonstrate compliance with the limits of 0.8 $(1.2\ S_h + S_A)$ or 1.8 S_h as applicable.

(g) The terminal ends of these piping runs will be considered to originate at a point adjacent to the restraints located inside and outside the containment which are:

(1) Located reasonably close to the isolation valves

(2) Capable of withstanding the loadings resulting from a postulated pipe rupture beyond this portion of the piping such that neither valve operability nor the leaktight integrity of the containment will be impaired.

3. Type of breaks in fluid system piping:

At locations where breaks are postulated to occur the following breaks are postulated:

a) Full cross sectional area circumferential breaks with at least 1 diameter pipe displacement in piping greater than one inch,

unless the separation is physically limited by piping restraint, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis. (This has been utilized in the reactor cavity design, as discussed in our meeting with you on 11/12/74).

(d) Single cross sectional area longitudinal breaks in piping four inches and greater. Longitudinal breaks need not be postulated at:

(1) The terminal ends provided the piping at the terminal ends contain no longitudinal welds; or

(2) Intermediate locations where the criterion for a minimum number of break locations must be satisfied.

Up to this point we have provided the details of criteria to determine the postulated break locations in the high energy piping. We would now like to discuss several points of clarification which we will use in applying this criteria in our design.

1. Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the pipe flow area exposed by the separation of the two sections of pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially and cause the pipe to move in the direction of the thrust force. Pipe whipping will be assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

2. For longitudinal breaks piping deflection will be assumed to occur in the direciton of the jet reaction unless limited by structural members, piping restraints, or piping stiffness as demonstrated by inelastic limit analysis.

3. Measures for restraint against pipe whipping as a result of the breaks postulated by the above criteria need not be provided for piping where any one of the following apply:

   a) When piping is physically separated (or isolated) from any essential safety related structure, system or component required to place the plant in a safe shutdown condition following the postulated rupture by protective barriers or is restrained from whipping by plant design features such as concrete encasement;

   b) When, following a single break, the unrestrained pipe movement of either end of the ruptured pipe cannot damage, to an unaccetable level, and essential safety related structure, system or component required to place the plant in a safe shutdown condition following the postulated rupture;

   c) When the energy associated with the whipping pipe can be demonstrated to be insufficient to impair, to an unacceptable level, the safety function of any essential safety related structure, system or component required to place the plant in a safe shutdown condition following the postulated rupture; (A whipping pipe is considered insufficient to rupture an impacted pipe of equal or larger nominal pipe size and equal or heavier wall thickness.)

4. For the purposes of calculating the stresses for ASME III, Code Class 1 piping, the operational plant conditions include:

   a) Plant operating conditions during reactor startup, operation at power, hot standby or reactor cooldown to the cold shutdown condition. The operational plant conditions do not include hydrostatic testing. However, it should be noted that the hydrostatic testing condition will be considered in calculating the cumulative usage factor.

b) Operational plant conditions include plant operating conditions during system transients that may occur with moderate frequency during plant service life and are anticipated operational occurences.

5. For the purposes of calculating the stresses for ASME III, Code Class 2 and 3 piping the system design condition will envelope the operational plant conditions (defined above for Code Class I components). Where the design condition results in an excessive number of postulated breaks, operational plant conditions defined consistent with those for Code Class 1 components will be utilized.

Enclosure No. 3

2

ADDITIONS (A)/EXCEPTIONS (E)/CLARIFICATIONS (C) TO
MR. GIAMBUSSO'S LETTER OF DECEMBER 1972

A/C/E	1972 LETTER POSITION/ FOOTNOTE	DESCRIPTION
E	1. The systems (or portions of systems) for which protection against pipe whip is required should be identified. Protection from pipe whip need not be provided if any of the following conditions will exist:	
	(a) Both of the following piping system conditions are met: (1) the service temperature is less than 200°F; and (2) the design pressure is 275 psig or less;	We propose to define high energy based on operating temperature and pressure vice service temperature and design pressure.
C	(b) The piping is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, or restrained features, such as concrete encasement.	We interpret the term "structures, systems or components important to safety" as follows: "any essential safety related structure system or component required to place the plant in a safe shutdown condition following the postulated rupture."
E	(c) Following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any structure, system, or component important to safety;	Pipe whip direction for circumferential breaks, will be determined by piping geometry and configuration and in the direction of the jet reaction force, vice in any direction about a plastic hinge formed at the nearest restraint.

A/C/E	1972 LETTER POSITION/ FOOTNOTE	DESCRIPTION
		b) We interpret the last part as follows: "can not damage, to an unacceptable level any essential safety related structure, system or component required to place the plant in a safe shutdown condition following the postulated rupture."
C	1.(d) The internal energy level associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component to an unacceptable level.	We interpret the last part of the position as follows: "impair, to an unacceptable level, the safety function of any essential safety related structure, system or component required to place the plant in a safe shutdown condition following the postulated rupture."
	2. The criteria used to determine the design basis piping break locations in the piping systems should be equivalent to the following: (a) ASME Section III Code Class I piping breaks should be postulated to occur at the following locations in each piping run or branch run: (1) Terminal Ends	

A/C/E	72 LETTER POSITION/ FOOTNOTE	DESCRIPTION
E	(2) any intermediate locations between terminal ends where the primary plus secondary stress intensities S_m (circumferential or longitudinal) derived on an elastically calculated basis under the loadings associated with one-half safe shutdown earthquake and operational plant conditions exceeds 2.0 S_m for ferritic steel, and 2.4 S_m for austenitic steel;	We propose to apply <u>2.4 Sm</u> for both ferritic and austenitic steel. This is reflected in your Mechanical Engineering Branch (MEB) position paper.
C	2.(a)(4) at intermediate locations in addition to those determined by (1) and (2), selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.	We propose to postulate intermediate breaks at locations (piping weld joints to fittings, valve or welded attachment) of <u>next highest stress until two intermediate break points are determined</u>. For piping runs shorter than 10 diameters only one intermediate break point will be postulated.
	2.(b) ASME Section III Code Class 2 and 3 piping breaks should be postulated to occur at the following locations in each piping run or branch run: (1) Terminal Ends	

A/C/E	72 LETTER POSITION/ FOOTNOTE	DESCRIPTION
E	(2) any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.9 ($S_h + S_A$) or the expansion stresses exceed 0.8 S_A; and	We propose to apply stress criteria of 0.8 ($1.2S_h + S_A$) vice 0.9 ($S_h + S_A$) and $0.8S_A$. This is consistent with the MEB position paper.
C	(3) intermediate locations in addition to these determined by (2) above, selected on reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.	We propose to postulate intermediate breaks at locations (piping weld joints to fittings, valve or welded attachments) of next highest stress until two intermediate break points are determined.
E	3. The criteria used to determine the pipe break orientation at the break locations as specified under 2 above should be equivalent to the following:	
	(a) Longitudinal breaks in piping runs and branch runs, 4 inches nominal pipe size and larger;	We will postulate longitudinal ruptures in piping - 4 inches nominal pipe size and larger except when: 1) the terminal ends contain no longitudinal welds; or

A/C/E	72 LETTER POSITION/ FOOTNOTE	DESCRIPTION
		2) <u>when intermediate locations are postulated to satisfy the criteria for a minimum number of breaks</u>.
C	Footnote 4. Operational plant conditions include normal reactor operation, upset conditions (e.g., anticipated operational occurrences), and testing conditions.	We interpret footnote 4, defining operational plant conditions, as it is used in Positions 2.(a)(2) and 2.(b)(2) as follows: 1) For the purposes of calculating the stresses for ASME III, Code Class 1 piping, the operational plant conditions include: a) Plant operating conditions during reactor startup, operation at power, hot standby or reactor cooldown to the cold shutdown condition; but not hydrostatic testing (the hydrostatic testing condition will be considered in calculating the cumulative usage factor).

A/C/E	1972 LETTER POSITION/ FOOTNOTE	DESCRIPTION
		b) Plant operating conditions during system trnasients that may occur with moderate frequency during plant service life and are anticipated operational occurences. 2) For the purposes of calculating the stresses for ASME III, Code Class 2 and 3 piping the system design condition will envelope the operational plant conditions (defined above for Code Class 1 components). As an alternate Where the design condition results in an excessive number of postulated breaks, operational plant conditions defined consistent with those for Code Class 1 components will be utilized.
E	Footnote 9. Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially and cause whipping in any direction normal to the pipe axis.	Pipe whip direction for circumferential breaks, will be determined by piping <u>geometry and configuration and in the direction of the jet reaction force</u>, vice <u>whipping in any direction normal to the pipe axis</u>.

A/C/E	1972 LETTER POSITION/ FOOTNOTE	DESCRIPTION
A	Not addressed	We propose that the criteria for break locations in non-nuclear high energy piping be as follows: 1) Terminal ends 2) At intermediate fittings, welded attachments and valves.
A	Not addressed.	The proposed criteria where breaks will not be postulated in containment penetration area, is as follows: 1) Stress $< .8\ (1.2\ S_h + S_A)$ 2) Stress $<1.8\ S_h$ within region for breaks outside of region.

ADDITIONS (A)/EXCEPTIONS (E)/CLARIFICATIONS (C) TO REGULATORY GUIDE 1.46

A/C/E	REGULATORY POSITION/ FOOTNOTE	DESCRIPTION
E	C.1. ASME Section III Code Class 1 piping breaks should be postulated to occur at the following locations in each piping run or branch run:	
	b. Any intermediate locations between terminal ends where the primary plus secondary stress intensities (circumferential or longitudinal) derived on an elastically calculated basis under the loadings associated with specified seismic events and operational plant conditions exceed 2 S_m for ferrite steel and 2.4 S_m for austenitic steel;	We propose to apply 2.4 Sm for both ferritic and austenitic steel. This is reflected in your Mechanical Engineering Branch (MEB) position papers.
C	d. At intermediate locations in addition to those determined by regulatory positions 1.b. and 1.c. selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.	We propose to postulate intermediate breaks at locations (piping weld joints to fittings, valve or welding attachment) of next highest stress until two intermediate break points are determined. For piping runs shorter than 10 diameters only one intermediate break point will be postulated.

A/C/E	REGULATORY POSITION/ FOOTNOTE	DESCRIPTION
E	C.2 ASME Section III Code Class 2 and 3 piping breaks should be postulated to occur at the following locations in each piping run or branch run:	
	b. Any intermediate locations between terminal ends where either the circumferential or logitudinal stresses derived on an elastically calculated basis under the loadings associated with specified seismic events and operational plant conditions exceed 0.8 ($S_h + S_A$)	We propose to apply stress criteria <u>of 0.8 (1.2 $S_h + S_A$)</u> <u>vice 0.8 ($S_h + S_A$)</u>. This is consistent with the MEB position paper.
C	C.2.d. Intermediate locations in addition to those determined by regulatory positions 2.b. and 2.c., selected on reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.	We propose to postulate intermediate breaks at locations (piping weld joints to fittings, valve or welded attachments) of <u>next highest stress until two intermediate break points are determined</u>.
E	C.3 The following types of breaks should be postulated at the locations identified by regulatory positions 1 and 2 above:	We will <u>postulate logitudinal ruptures</u> in piping - 4 inches nominal pipe size and larger except when:

A/C/E	REGULATORY POSITION/ FOOTNOTE	
	a. Longitudinal breaks in piping runs and branch runs 4 inches nominal pipe size and larger;	1) the terminal ends contain no longitudinal welds; or 2) when intermediate locations are postulated just to satisfy the criteria for a minimum number of breaks. In this case only circumferential breaks will be postulated.
C	C.4 Measures for restraint against pipe whipping as a result of the design basis breaks postulated to occur at the locations specified under regulatory positions 1 and 2 above need not be provided for piping where any one of the following applies:	
	a. The piping is physically separated (or isolated) from other piping or components by protective barriers or is restrained from whipping by plant design features such as concrete encasement;	We interpret the term "other piping or components" as follows: "any essential safety related structure, system or component required to place the plant in a safe shutdown condition following the postulated rupture."
E	b. Following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any direction about a plastic hinge formed at the nearest pipe whip restraint cannot damage any structure, system, or component important to safety;	a) Pipe whip direction for circumferential breaks, will be determined by piping geometry and configuration and in the direction of the jet reaction force, vice in any direction about a plastic hinge formed at the nearest restraint.

A/C/E	REGULATORY POSITION/ FOOTNOTE	DESCRIPTION
		b) We interpret the last part as follows: "cannot damage, to an unacceptable level any essential safety related structure, system or component required to place the plant in a safe shutdown condition following the postulated rupture".
C	C.4.c. The energy associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component important to safety to an unacceptable level;	a) We interpret the last part of the position as follows: "impair, to an unacceptable level, the safety function of any essential safety related structure, system or component required to place the plant in a safety shutdown condition following the postulated rupture."
E	C.4.d. Both of the following piping system conditions are met: (1) the design temperature is 200°F or less, and (2) the design pressure is 275 psig or less.	We propose to define high energy based on operating temperature and pressure, not design temperature and pressure.

A/C/E	REGULATORY POSITION/ FOOTNOTE	DESCRIPTION
E/C	Footnote 5. Operational plant conditions include normal reactor operation, upset conditions (e.g., anticipated operational occurrences), and testing conditions	We interpret footnote 5, defining operational plant conditions, as it is used in Regulatory Positions C.1.b, C.1.c and C.2.b as follows: 1) For the purposes of calculating the stresses for ASME III, Code Class I piping, the operational plant conditions include: a) Plant operating conditions during reactor startup, operation at power, hot standby or reactor cooldown to the cold shutdown condition; but not hydrostatic testing (the hydrostatic testing condition will be considered in calculating the cumulative usage factor). b) Plant operating conditions during system transients that may occur with moderate frequency during plant service life and are anticipated operational occurrences.

A/C/E	REGULATORY POSITION/ FOOTNOTE	DESCRIPTION
		2) For the purposes of calculating the stresses for ASME III, Code Class 2 and 3 piping the system design condition will envelope the operational plant conditions (defined above for Code Class I components) Where the design condition results in an excessive number of postulated breaks, operational plant conditions defined consistent with those for code class I components will be utilized.
E	Footnote 10. Longitudinal breaks are parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the sum of the effective cross-sectional flow area upstream of the break location and downstream of the break location or is equal to a break area determined by test data which defines the break geometry. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.	We propose to take single cross sectional <u>area longitudinal breaks only</u>.

A/C/E	REGULATORY POSITION/ FOOTNOTE	DESCRIPTION
E	Footnote 11. Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially and cause whipping in any direction normal to the pipe axis.	Pipe whip direction for circumferential breaks, will be determined by piping geometry and configuration and in the direction of the jet reaction force, vice whipping in any direction normal to the pipe axis.
C	Footnote 14. A whipping pipe should be considered as sufficient to rupture an impacted pipe of smaller nominal pipe size and lighter wall thickness.	We interpret this as follows: "A whipping pipe is considered insufficient to rupture an impacted pipe of equal or larger nominal pipe size and equal or heavier wall thickness."

A/C/E	REGULATORY POSITION/ FOOTNOTE	DESCRIPTION
A	Not addressed.	We propose that the criteria for break locations in non-nuclear high energy piping be as follows: 1) terminal ends 2) at intermediate fittings, welded attachments and valves.
A	Not addressed.	The proposed criteria where breaks will not be postulated in containment penetration area, is as follows: 1) Stress $< .8\ (1.2\ S_h + S_A)$ 2) Stress $< 1.8\ S_h$ within region for breaks outside of region.

361/362

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EP Project Manager - R. Froelich
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J. M. Kovacs
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H. L. Brammer
G. C. Millman

FEB 12 1975

DOCKET NOS: 50-361 and 50-362

APPLICANT: Southern California Edison Company (SCE)

FACILITY: San Onofre Nuclear Generating Station, Units 2 and 3

SUMMARY OF JANUARY 30, 1975 MEETING WITH SCE TO DISCUSS CONFORMANCE WITH RECOMMENDATIONS OF REGULATORY GUIDE 1.18

On January 30, 1975, representatives of SCE and Bechtel met with the Regulatory staff in Bethesda. SCE had requested the meeting to discuss the degree of conformance with the recommendations of Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments". A list of attendees is contained in Enclosure No. 1.

SCE stated that the San Onofre Units 2 and 3 PSAR had committed to comply with Regulatory Guide 1.18. SCE explained that in the final design of the containment dome, a hemispherical dome was used instead of the original ellipsoidal configuration. Due to construction and schedule delays incurred by other applicants, the San Onofre Units 2 and 3 containment design has become a prototype for purposes of complying with Regulatory Guide 1.18. SCE stated that they were proposing a technique for meeting the recommendations of Regulatory Guide 1.18 which would provide demonstration of acceptability under Regulatory Guide 1.18 only for features unique to the San Onofre Units 2 and 3 containment design.

Bechtel described the design of the containment structure. The containment structure consists of a prestressed post-tensioned concrete vessel in the shape of a right vertical cylinder with a hemispherical dome and a flat, conventionally reinforced basemat. Bechtel then described the post-tensioning system.

Bechtel then identified as prototypical the following features of the San Onofre Units 2 and 3 containment structural design per Appendix A of Regulatory Guide 1.18:

(1) Number of buttresses
(2) Buttresses in the dome
(3) Geometry of tendons in the dome
(4) Shape of dome
(5) Tendon size

Memo (4)

Each of the features identified above were then discussed (see Enclosure No. 2).

The extent of the instrumentation to be provided for SIT tests was discussed. SCE agreed to submit details of the proposed program with technical justification for the staff's review. The staff agreed to evaluate the proposed program for compliance with Regulatory Guide 1.18 in a timely manner upon submittal of the detailed information and to inform SCE of the results.

Original Signed by
Patrick D. O'Reilly

Patrick D. O'Reilly
Light Water Reactors
Project Branch 1-3
Division of Reactor Licensing

Enclosures:
1. Attendance List
2. Compliance with
   Regulatory Guide 1.18
   San Onofre Units 2 & 3

cc: Southern California Edison Company
Rollin E. Woodburg
Chickering & Gregory
Mr. Larry E. Moss
Mr. David Sakai
Frederick P. Sutherland
Mr. Kenneth E. Carr
Alan R. Watts
Lawrence Q. Garcia
George Spiegel

OFFICE➤	RL:LWR 1-3					p
SURNAME➤	PDO'Reilly:cls					
DATE➤	2/ /75					

ENCLOSURE NO. 1

ATTENDANCE LIST

Nuclear Regulatory Commission

I. Villalva
D. C. Jeng
P. D. O'Reilly

Southern California Edison Co.

J. H. Roecker
D. E. Nunn
W. C. Moody
D. F. Martin

Bechtel

T. H. Stick
O. S. Welty
H. T. Hill
H. R. Reuter
L. G. Hersh

ENCLOSURE NO. 2

COMPLIANCE WITH
REGULATORY GUIDE 1.18
SAN ONOFRE UNITS 2&3

1. Scope

Regulatory Guide 1.18 establishes criteria for a pre-operational structural integrity test wherein quantitative information is obtained concerning the structural response of the containment structure when subjected to internal pressurization.

2. Position

The Applicant intends to comply with Regulatory Guide 1.18. Specifically with respect to paragraph C.5 we will conduct prototype testing as needed to demonstrate the acceptability of new or unusual features incorporated into the design of the containment structure.

3. Description of Structure

The containment structure is a prestressed, post-tensioned concrete vessel in the shape of a right vertical cylinder with a hemispherical dome and a flat, conventionally reinforced basemat.

The post-tensioning incorporates a strand system with an approximate capacity of 1,000 tons per tendon.

The hoop tendons are anchored along three vertical buttresses and extend up into the dome to an axis of 45^{o} above the transition plane.

The remaining tendons, inverted U-shaped tendons, are divided into two separate groups oriented at ~~180°~~ 90 to each other. All inverted U-shaped tendons within a specific group are positioned in parallel vertical planes with both end anchorages terminating in the basemat.

4. <u>Prototypical Features Per Appendix A of Regulatory Guide 1.18</u>

   A. Number of buttresses other than 6

   B. Buttresses in the dome

   C. Geometry of tendons in the dome

   D. Shape of dome

   G. Tendon Size

   Note: All items except A and G are a direct result of the change in the shape of the dome and therefore will be discussed as a single item.

5. <u>Number of Buttresses</u>

   The influence of a buttress is a localized effect. Prototype tests of previous structures show little variation in membrane conditions and the variation in response at the discontinuity regions would be anticipated.

   A three buttress arrangement provides a more uniform stress distribution from the post-tensioning system.

   Displacement responses from SIT tests for three buttress containment structures have shown good correlation between predicted responses and measured responses.

6. Size of Tendon

The difference in tendon size affects only the end anchorage region of the structure. Full scale tests of end anchorages and a buttress zone have been carried out using the large capacity tendons. These tests have been documented in Bechtel Topical Reports BC-TOP-7 and 8 and have been approved by the AEC.

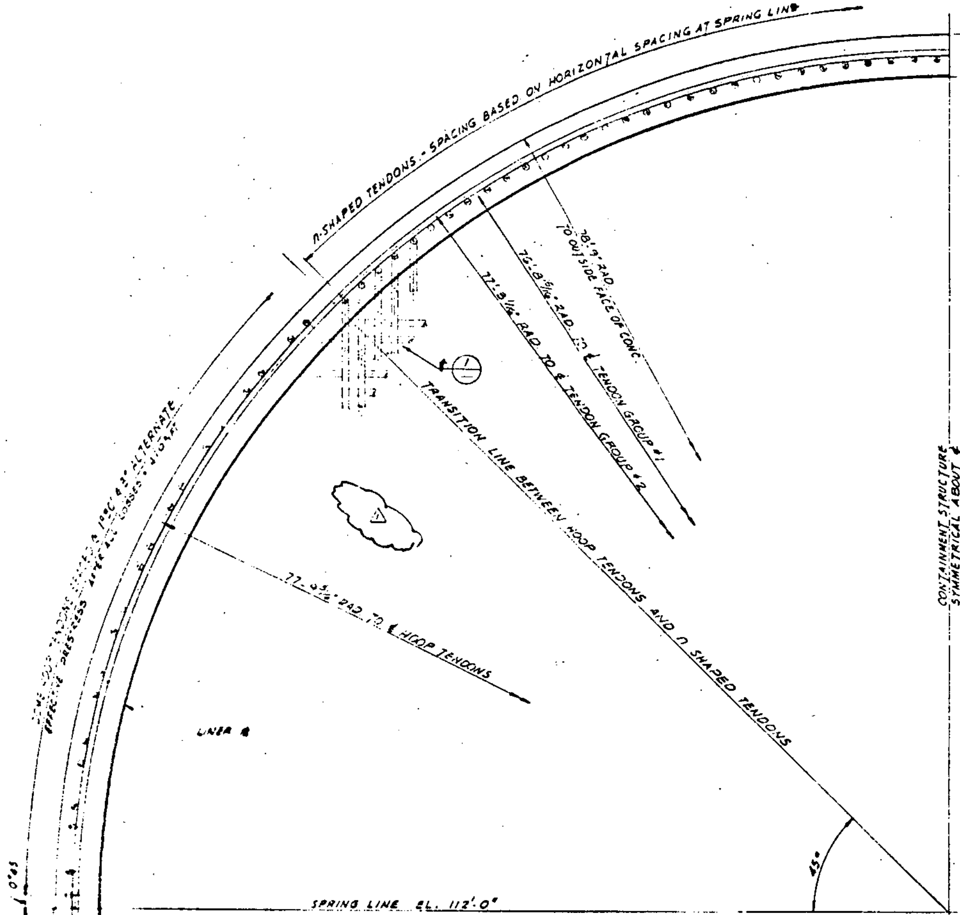
7. Dome Configuration and Associated Modifications

The results of the change in dome configuration affects only the dome region and a small portion of the upper wall just below the transition point. The proposed prototype testing will be limited to this region. Strain measurements will be taken along two meridian lines; one located along a buttress and the other midway between buttresses. Strain measurements will be monitored 20 feet below the springline, at the springline, 45° above the springline (termination of the hoop tendons) and at the apex of the dome. In addition, strain measurements will be taken on the mid-buttress meridian at a point were the influence of the three tendon pattern can be evaluated.

8. Extent of Instrumentation

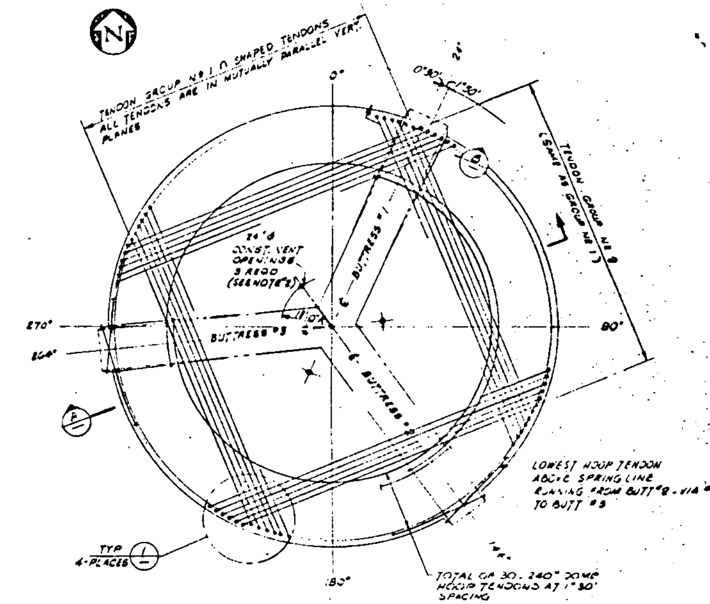
Analytical methods are based upon linear elastic, thin shell, small displacement theory which assumes a linear variation of strain with depth. Previous SIT tests have demonstrated good correlation between predicted responses and measured responses. Therefore strain measurements need only be taken at two positions within a cross-section since a third measurement would be redundant. Strain gages will be located at the inside face and at the outside face of each position identified

in Paragraph 7. In addition, concrete temperatures will also be measured at several of the above locations to correlate thermal effects on the response of the structure.



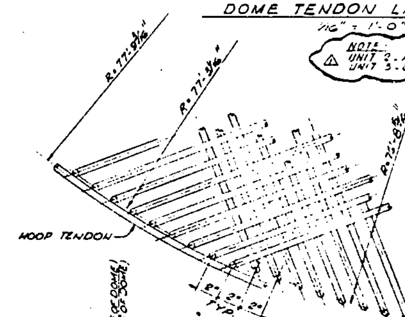
NOTE:
DOME HOOP TENDONS SHALL BE ANCHORED AT DOME BUTTRESSES IN SAME FASHION & CONTINUING SEQUENCE AS WALL HOOP TENDONS.

SECTION (A)
1/8"=1'-0"

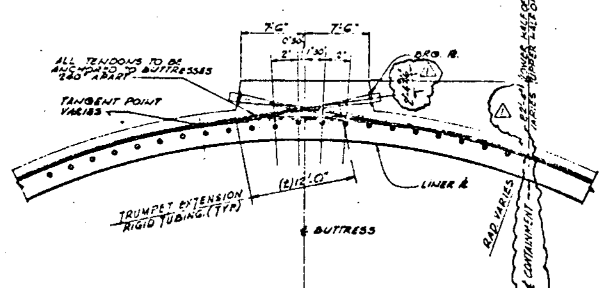


DOME TENDON LAYOUT PLAN
1/16" = 1'-0"

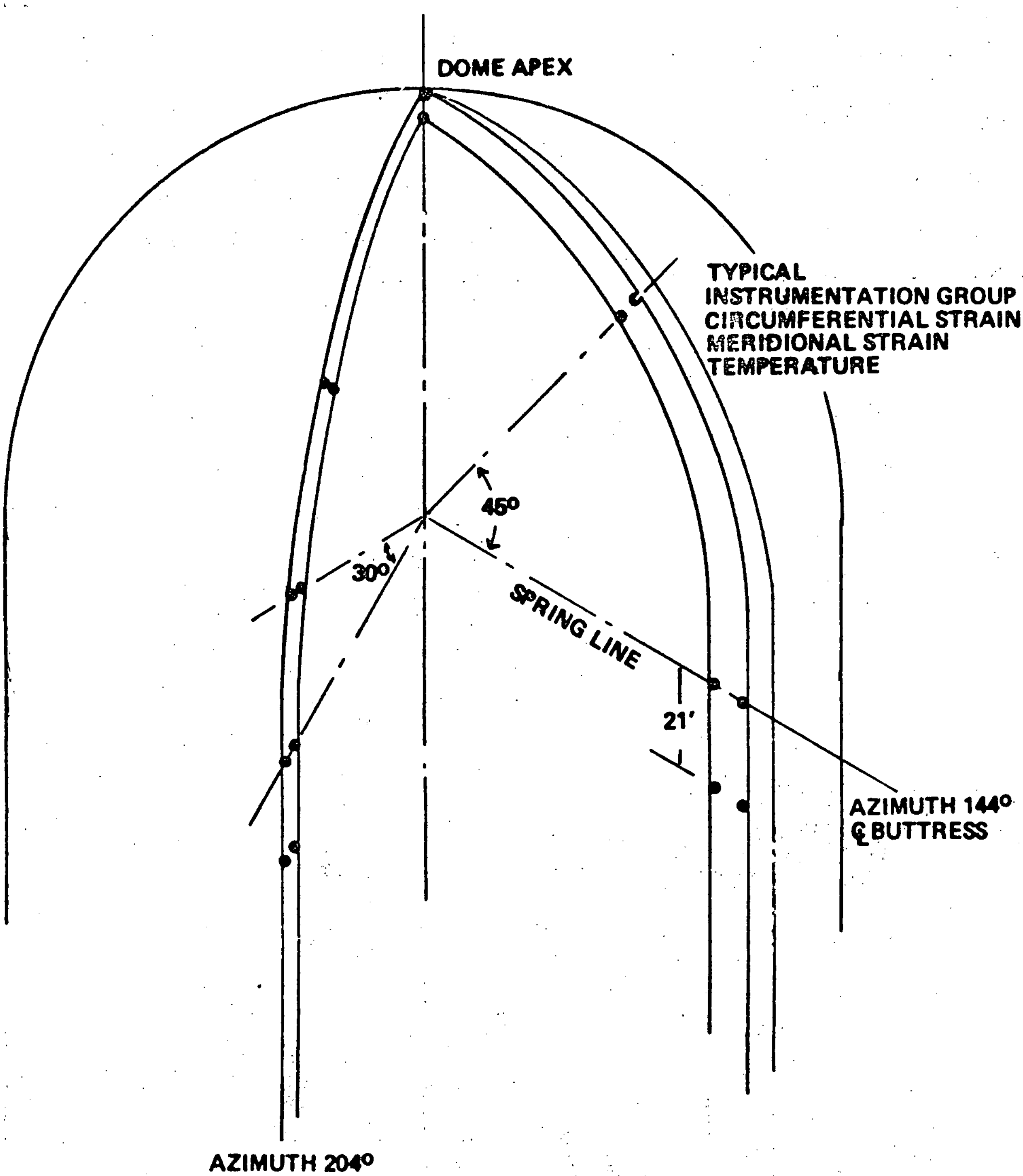
NOTE:
UNIT 2 - AS SHOWN
UNIT 3 - OPP. HAND ABOUT EAST-WEST ℄



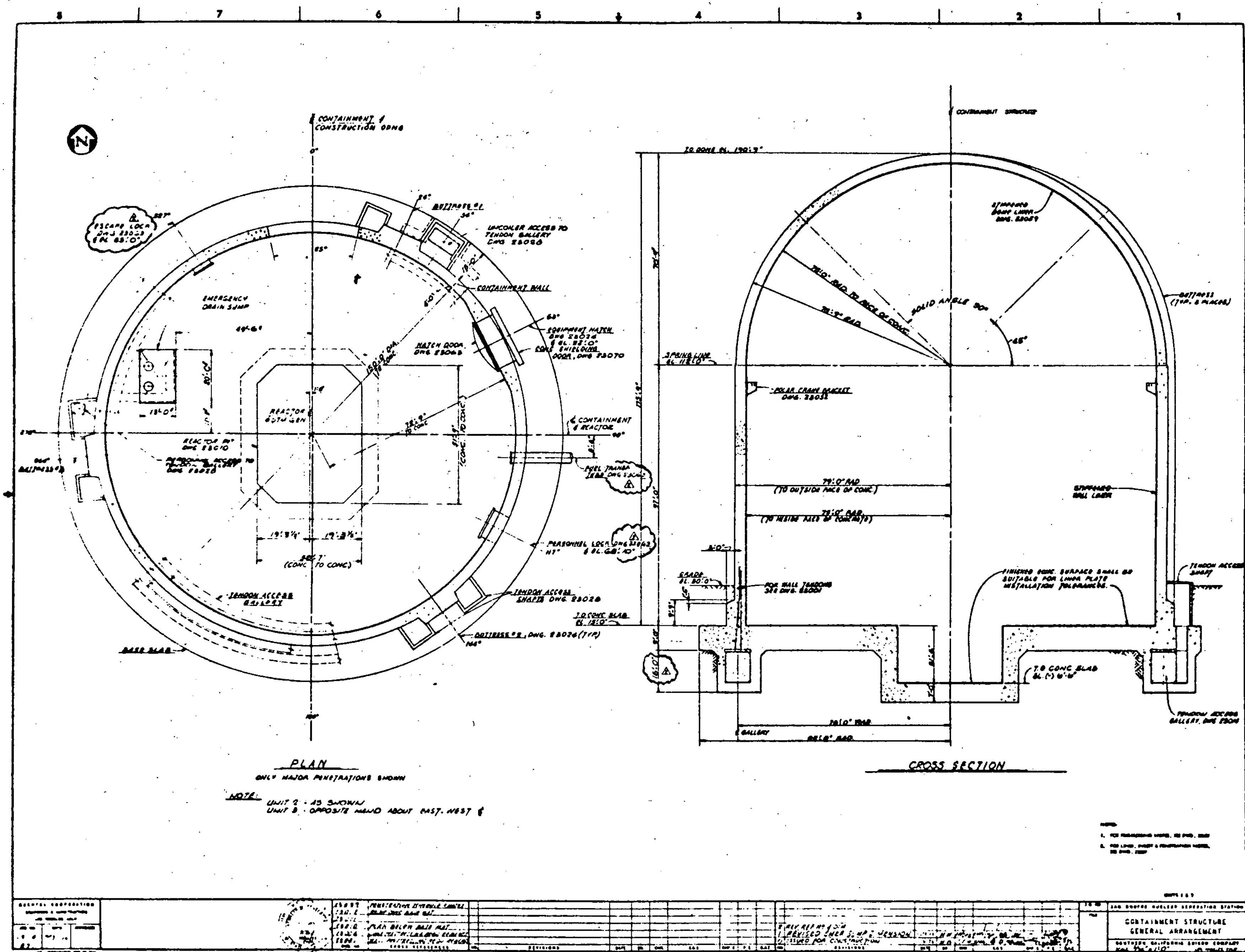
DETAIL (1)
N.T.S.



BUTTRESS SECTION AT DOME (B)
FROM SPRING LINE TO 45° VERT.
3/16" = 1'-0"



CUTAWAY VIEW OF CONTAINMENT
SHOWING INSTRUMENTATION LOCATIONS



CONTAINMENT ℄ CONSTRUCTION DRWG
ESCAPE LOCK DWG 23063 ℄ EL 65'-0"
BUTTRESS #1
UNCOILER ACCESS TO TENDON GALLERY DWG 23026
CONTAINMENT WALL
EMERGENCY DRAIN SUMP
EQUIPMENT HATCH DWG 23074 ℄ EL 55'-0"
HATCH DOOR DWG 23066
CONC SHIELDING DOOR DWG 23070
150'-0" DIA. TO CONC.
REACTOR & STM GEN
℄ CONTAINMENT ℄ REACTOR
PERSONNEL ACCESS TO TENDON GALLERY DWG 23026
FUEL TRANSFER TUBE
PERSONNEL LOCK DWG 23062 ℄ EL 68'-10"
TENDON ACCESS GALLERY
TENDON ACCESS SHAFTS DWG 23026
BUTTRESS #2, DWG 23026 (TYP)
BASE SLAB
PLAN
ONLY MAJOR PENETRATIONS SHOWN
NOTE: UNIT 2 - AS SHOWN
UNIT 3 - OPPOSITE HAND ABOUT EAST-WEST ℄
℄ CONTAINMENT STRUCTURE
TO DOME EL 190'-9"
STIFFENED DOME LINER DWG 23059
70'-0" RAD. TO FACE OF CONC.
75'-0" RAD
SOLID ANGLE 90°
45°
BUTTRESS (TYP. 3 PLACES)
SPRING LINE EL 115'-0"
POLAR CRANE BRACKET DWG 23052
79'-0" RAD (TO OUTSIDE FACE OF CONC.)
75'-0" RAD. (TO INSIDE FACE OF CONC.)
STIFFENED WALL LINER
FINISHED CONC. SURFACE SHALL BE SUITABLE FOR LINER PLATE INSTALLATION TOLERANCES.
GRADE EL 30'-0"
FOR WALL TENDONS SEE DWG 23001
TO CONC SLAB EL 15'-0"
TENDON ACCESS SHAFT
TO CONC SLAB
TENDON ACCESS GALLERY, DWG 23026
℄ GALLERY
CROSS SECTION
CONTAINMENT STRUCTURE GENERAL ARRANGEMENT
SAN ONOFRE NUCLEAR GENERATING STATION
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BECHTEL CORPORATION
23000

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Docket Nos. 50-361
and 50-362

JAN. 27 1975

R. F. Fraley
Executive Secretary
Advisory Committee on
Reactor Safeguards

REGULATORY STAFF'S EVALUATION OF SIGNIFICANCE OF RECENT SEISMIC ACTIVITY IN SOUTHERN CALIFORNIA NEAR THE CRISTIANITOS FAULT

In your January 16, 1975 memo to E. G. Case, you expressed interest in the staff's evaluation of recent seismic activity in southern California that may be associated with the Cristianitos Fault. The staff is currently evaluating the effect of this recent activity on the conclusions regarding the San Onofre Nuclear Generating Station site that were stated in the Safety Evaluation Report for Units 2 and 3.

We will be prepared to report on the status of our review at the next meeting of the Committee, scheduled February 6 - 8, 1975. This matter was discussed previously with T. G. McCreless of your staff on January 20, 1975.

Original signed by R. C. DeYoung

R. C. DeYoung, Assistant Director
for Light Water Reactors, Group 1
Division of Reactor Licensing

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SURNAME	PO'Reilly:pg	ODParr	HRDenton	RCDeYoung		
DATE	1/23/75	1/23/75	1/[illegible]/75	1/27/75		

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F. Schroeder, Acting Director, Division of Technical Review
THRU: R. C. DeYoung, Assistant Director for Light Water Reactors Group 1

TECHNICAL ASSISTANCE REQUEST - SAN ONOFRE

A need has arisen to reexamine the emergency evacuation requirements associated with the exclusion area and the immediate environs of the San Onofre site. To permit this to be done we request your technical assistance in providing us with the following information:

(1) For San Onofre Units 2 and 3

Estimated thyroid and whole body doses as a function of time out to two hours after a LOCA, and as a function of distance from the nearest exclusion area land boundary to the most distant exclusion area land boundary. The appropriate meteorology for the plume direction (toward the sea or toward the land) should be used. The assumptions used should include a core power level of 3390 MWt, the assumptions of Regulatory Guide 1.4, a containment leak rate of 0.3% per day, the spray removal constants appropriate to the design and any other assumptions appropriate to the plant and the site including, for example, factors to account for topographic irregularities particularly in the direction towards the ocean.

We request that, in addition to the dose estimates for Regulatory Guide 1.4 radioactivity releases from the fuel to the containment (instantaneous release of 25% of the radioiodines and 100% of the noble gases) dose estimates for the following releases also be provided:

(a) The release assuming all ESF perform as designed.

(b) The releases assuming the ESF perform in a degraded manner such that 10% of the core becomes molten and releases its activity to the containment at a linear rate beginning 5 minutes after the LOCA occurs and ending 35 minutes after the occurrence.

memo 4

We also request that a calculation of dose concentration within the plume cross section at 100 m. from Units 2 and 3 as a function of time be provided using the most conservative set of conditions assumed in the above analyses.

(2) For San Onofre 1

The same type of estimates as for Units 2 and 3 except that the core power level should be 1347 MWt, the containment leak rate should be 0.1% per day, and other plant design and site factors appropriate to the unit should be used in the calculations. The dose concentration within the plume cross section at 75 meters from Unit 1 should be calculated using the most conservative set of conditions used in the previous analyses.

This information is needed as soon as possible. The Accident Analyses Branch has been developing some of the needed information for the past several days. We request that the information be provided to us by January 31, 1975. We are prepared to meet with Site Safety representatives to clarify our needs.

Original Signed by
Olan Parr

Olan D. Parr, Chief
Light Water Reactors
Project Branch 1-3
Division of Reactor Licensing

cc: A. Giambusso
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P. O'Reilly
D. Eisenhut
D. McDonald
E. Case
R. Klecker
T. Carter
A. Burger
R. Purple
D. Kartalie
L. Chandler
D. Skovholt
W. Houston

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JAN. 28 1975

Docket Nos. STN 50-206
STN 50-361
and STN 50-362

D. J. Skovholt, Assistant Director for Quality Assurance & Operations
THRU: R. C. DeYoung, Assistant Director for Light Water Reactors, Group 1

Original Signed by R. C. DeYoung

REQUEST FOR REEVALUATION OF EVACUATION REQUIREMENTS FOR SAN ONOFRE SITE

In your April 23, 1974 response to TAR-769, you stated that the conditions accepted by the applicant from the California Coastal Zone Conservation Commission did not alter your conclusions about the acceptability of the plans for coping with emergencies at the San Onofre site as stated in the Safety Evaluation Report for Units 2 and 3. At our request, you reviewed the supplemental memorandum concerning proposed recreational facilities in the site vicinity that was submitted to the Appeal Board by the applicants on June 13, 1974. The J. R. Sears' memo of August 14, 1974 reconfirms that the staff's position regarding evacuation of the state park areas had not changed.

Recently a need has arisen to reexamine the emergency evacuation requirements associated with the exclusion area and the immediate environs of the San Onofre site. We request your technical assistance in providing us with an evaluation of the feasibility of the evacuation of the exclusion area and immediate environs of the San Onofre site under 10 CFR Part 100 guidelines. The information that you will require for performing this evaluation is currently being developed by the Accident Analysis Branch, TR (See attached memo for details).

This evaluation is needed as soon as possible. We request that the evaluation be provided to us by January 31, 1975. Although this schedule is the same as the schedule for development of input which you require, some of the necessary information has already been generated by TR. We are prepared to meet with your staff to clarify our needs.

Original Signed by
O. D. Parr

Olan D. Parr, Chief
Light Water Reactors
Project Branch 1-3
Division of Reactor Licensing

memo

OFFICE	RL:LWR 1-3 PDO	RL:LWR 1-3 ODP	RL:AD/LWR 1			
SURNAME	PO'Reilly:pg	ODParr	RCDeYoung			
DATE	1/23/75	1/28/75	1/28/75			

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

Docket Nos. 50-206
50-361
and 50-362

JAN 23 1975

F. Schroeder, Acting Director, Division of Technical Review
THRU: R. C. DeYoung, Assistant Director for Light Water Reactors Group RCD

TECHNICAL ASSISTANCE REQUEST - SAN ONOFRE

A need has arisen to reexamine the emergency evacuation requirements associated with the exclusion area and the immediate environs of the San Onofre site. To permit this to be done we request your technical assistance in providing us with the following information:

(1) <u>For San Onofre Units 2 and 3</u>

Estimated thyroid and whole body doses as a function of time out to two hours after a LOCA, and as a function of distance from the nearest exclusion area land boundary to the most distant exclusion area land boundary. The appropriate meteorology for the plume direction (toward the sea or toward the land) should be used. The assumptions used should include a core power level of 3390 MWt, the assumptions of Regulatory Guide 1.4, a containment leak rate of 0.3% per day, the spray removal constants appropriate to the design and any other assumptions appropriate to the plant and the site including, for example, factors to account for topographic irregularities particularly in the direction towards the ocean.

We request that, in addition to the dose estimates for Regulatory Guide 1.4 radioactivity releases from the fuel to the containment (instantaneous release of 25% of the radioiodines and 100% of the noble gases) dose estimates for the following releases also be provided:

(a) The release assuming all ESF perform as designed.

(b) The releases assuming the ESF perform in a degraded manner such that 10% of the core becomes molten and releases its activity to the containment at a linear rate beginning 5 minutes after the LOCA occurs and ending 35 minutes after the occurrence.

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We also request that a calculation of dose concentration within the plume cross section at 100 m. from Units 2 and 3 as a function of time be provided using the most conservative set of conditions assumed in the above analyses.

(2) For San Onofre 1

The same type of estimates as for Units 2 and 3 except that the core power level should be 1347 MWt, the containment leak rate should be 0.1% per day, and other plant design and site factors appropriate to the unit should be used in the calculations. The dose concentration within the plume cross section at 75 meters from Unit 1 should be calculated using the most conservative set of conditions used in the previous analyses.

This information is needed as soon as possible. The Accident Analyses Branch has been developing some of the needed information for the past several days. We request that the information be provided to us by January 31, 1975. We are prepared to meet with Site Safety representatives to clarify our needs.

Olan D. Parr

Olan D. Parr, Chief
Light Water Reactors
Project Branch 1-3
Division of Reactor Licensing

cc: A. Giambusso
H. Denton
B. Grimes
K. Goller
O. Parr
P. O'Reilly
D. Eisenhut
D. McDonald
E. Case
R. Klecker
T. Carter
A. Burger
R. Purple
D. Kartalia
L. Chandler
D. Skovholt
W. Houston

Docket Nos. 50-361 ✓
and 50-362 ✓

JAN 13 1975

Olan D. Parr, Chief, Light Water Reactors Project Branch 1-3, L

NOTICE OF MEETING WITH SOUTHERN CALIFORNIA EDISON COMPANY - SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

Time & Date: 9:00 a.m.
January 30, 1975

Location: Room P-110
Bethesda, Maryland

Purpose: To discuss the design basis criteria for pipe whip for San Onofre 2 and 3.

Participants: SOUTHERN CALIFORNIA EDISON COMPANY
(H. Ray, W. Moody)

BECHTEL
(T. Stick)

COMBUSTION ENGINEERING
(R. Tease)

REGULATORY STAFF
(R. Bosnak, J. Kovacs, P. O'Reilly)

Original Signed by
Patrick D. O'Reilly

Patrick D. O'Reilly
Light Water Reactors
Project Branch 1-3
Directorate of Licensing

cc: see attached list

OFFICE ➤	L:LWR 1-3 PDO					Mems
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JAN 14 1975

Docket Nos. 50-361/362

David E. Kartalia, Assistant Chief Hearing Counsel
THRU: Donald J. Skovholt, Assistant Director for Quality Assurance and Operations, L

Original signed by
Donald J. Skovholt

STAFF REVIEW OF ALAB-248 RE SAN ONOFRE 2, 3; SITE SECURITY ISSUE

Reference: Note to D. Skovholt, dated January 6, 1975

The decision of ALAB-248 has raised the question of plant security for the San Onofre reactor site in that overnight camping will be permitted within the exclusion area directly next to the station boundary. Drawing Number 2, attached to the applicant's submittal to the ALAB dated June 13, 1974, and entitled Supplemental Memorandum Concerning Proposed Recreational Facilities in the Immediate Vicinity of Proposed Unit Nos. 2 and 3 of the San Onofre Nuclear Generating Station, illustrates where such camping may be located. Tent camping is proposed on the bluff to the northwest of the reactor site adjacent to the station boundary.

The applicant has described his preliminary plans for site security in Section 12.3.4 of the PSAR. The staff's evaluation of these preliminary plans is in Section 3.4 of the Safety Evaluation Report, dated October 20, 1972. Pursuant to your request, we have re-evaluated these preliminary plans on the basis of acceptance criteria in Regulatory Guide 1.17, Protection of Nuclear Power Plants Against Industrial Sabotage, June 1973, which endorses the requirements and recommendations of American National Standards Institute ANSI N18.17, Industrial Security for Nuclear Power Plants, dated March 23, 1973.

ANSI N18.17 designates three security areas, each within another. Vital areas contain vital equipment and are housed in structures protected by security patrol, operator surveillance, lock and key, and intrusion detection devices. Vital areas are within protected areas. Protected areas are enclosed by a physical barrier, either natural or a fence, usually around the plant buildings with a clear area on both sides of the barrier to afford a clear field of vision. The applicant has indicated that an eight-foot chain link fence will surround the protected area in conformance with the ANSI standard. The protected area barrier must be monitored regularly by security patrols. Only authorized personnel are permitted access to protected areas. The protected area in turn, is within the owner-controlled area. The owner-controlled area marks the boundary of the owner's property and is marked by signs or other means to designate private property. The owner-controlled area is not considered a sensitive security area and it is not necessarily enclosed by fencing.

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The applicant in this case, however, plans to surround the plant boundary, which corresponds to the ANSI N18.17 owner-controlled area, with another eight-foot chain link fence with normal access through a single entry gate controlled by a security officer on a 24-hour basis, with periodic security patrols of the entire property. Consequently, it is the staff judgment that the owner-controlled area is sufficiently protected to deter unauthorized access from the camping area next to the site boundary.

Original Signed by
R. Wayne Houston

R. Wayne Houston, Chief
Industrial Security and
Emergency Planning Branch
Directorate of Licensing

cc: P. D. O'Reilly

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