September 7, 1972

50-361

Richard C. DeYoung, Assistant Director for Pressurized Water Reactors, L

TECHNICAL ASSISTANCE REQUEST - CONTAINMENT DESIGN PRESSURE

Your memo dated September 7, 1972 requested summary statements on the adequacy of containment design pressure for inclusion in the Safety Evaluations for Forked River 1, San Onofre 2 and 3, Zion 1 and 2 and Kewaunee by a target date of September 11, 1972. We do plan to provide assistance in this matter but it is obvious that a completion date of September 11, 1972 is not possible. I am advised by Knuth and Tedesco that additional information will be needed from the applicant. We will review the scope of work needed and our manpower resources and inform you early next week of our proposed schedule.

Original Signed by
Frank Schroeder, Jr.

Frank Schroeder, Assistant Deputy
Director for Technical Review
Directorate of Licensing

cc: J. M. Hendrie, L
A. Giambusso, L
W. G. McDonald, L
D. F. Knuth, L
R. L. Tedesco, L

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L-Suppl
L-Rdg

OFFICE ▶	L:TR					Memo
SURNAME ▶	FSchroeder:eaj					
DATE ▶	9/7/72					

Form AEC-318 (Rev. 9-53) AECM 0240 GPO 643—10—81465-1 445-078

Docket No. 50-361

SEP 7 1972

J. M. Hendrie, Deputy Director for Technical Review
Directorate of Licensing

TECHNICAL ASSISTANCE REQUEST - CONTAINMENT DESIGN PRESSURE

Forked River 1	CP	Docket No. 50-363	P.M. - C. Moon
San Onofre 2/3	CP	Docket No. 50-361/362	P.M. - R. Birkel
Zion 1/2	OL	Docket No. 50-295/304	P.M. - R. Birkel
Kewaunee	OL	Docket No. 50-305	P.M. - L. Crocker

Requesting Branch: PWR Branch No. 3 & No. 2 (for Kewaunee)
Project Manager: Indicated above
Technical Review Branches: Reactor Systems and Containment Systems

Description of Request

The ACRS reports on San Onofre 2/3, dated July 21, 1972, and on Forked River 1, dated August 17, 1972, included the statement, "The Committee understands that the Regulatory Staff is reviewing the adequacy of the proposed design pressure for the reactor containment building. The Committee wishes to be kept informed." The ACRS reports on Zion 1/2 and Kewaunee, both dated August 17, 1972, included the statement, "The Committee recommends that the Regulatory Staff confirm the adequacy of the applicant's analysis of peak overall accident pressures during postulated loss-of-coolant accidents, as well as the response of compartment walls within the containment to dynamic forces during such events."

Please provide summary statements on this matter for inclusion in the Safety Evaluations for the listed facilities. The reactor vendors and A-E's are (1) Combustion Engineering, Inc. and Burns & Roe, Inc. for Forked River, (2) Combustion Engineering, Inc. and Bechtel Corporation for San Onofre 2/3, (3) Westinghouse Electric Corporation and Sargent and Lundy for Zion 1/2, and (4) Westinghouse Electric Corporation and Pioneer Service and Engineering Company for Kewaunee.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Target Date for Completion

The target dates for issuance of our Safety Evaluations for these applications were all prior to September 1, 1972. Please advise if the requested information cannot be provided by a target date of September 11, 1972.

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

cc: R. S. Boyd, L
D. J. Skovholt, L
D. R. Muller, L
H. Denton, L
W. McDonald, L
R. W. Klecker, L
M. Rosen, L
R. R. Maccary, L
D. F. Knuth, L
R. L. Tedesco, L
V. Stello, L
G. Lainas, L
A. Schwencer, L
K. Kniel, L
D. Vassallo, L
K. R. Goller, L
C. W. Moon, L

DISTRIBUTION:
Docket (6)
RP Reading
PWR-3 Reading
RABirkel, L
LCrocker, L

OFFICE ▶	L:PWR-3	L:PWR-3	L:AD/PWRs		
SURNAME ▶	x7415 CWM CWMoon:esp	KRG KRGoller	RCDeYoung		
DATE ▶	9/6/72	9/6/72	9/7/72		

SEP 1 1972

R. C. DeYoung, Assistant Director for PWR's, L

SAN ONOFRE 2/3

PLANT NAME: San Onofre 2/3
LICENSING STAGE: CP
DOCKET NUMBER: 50-361, 362
RESPONSIBLE BRANCH: PWR #3
REQUESTED COMPLETION DATE: NA
APPLICANTS RESPONSE DATE NECESSARY FOR
NEXT ACTION PLANNED ON PROJECT: NA
DESCRIPTION OF RESPONSE: NA
REVIEW STATUS: Complete Input For Public Safety
Analysis Report

Attached as requested is a correction sheet for the design basis accident doses for San Onofre 2/3. These doses are based on the use of Pasquill type "E" meteorology conditions with a wind speed of two meters/sec and a containment building leak rate of 0.3%/day.

Original signed by
H. R. Denton
Harold R. Denton, Assistant Director
for Site Safety
Directorate of Licensing

Enclosure:
As stated

cc: w/o enclosure
A. Giambusso
W. McDonald

cc: w/enclosure
S. H. Hanauer
J. Hendrie
W. P. Gammill
K. Goller
R. Birkel
C. Ferrell

DISTRIBUTION:
Docket File - 50-361, 362
L:Rdg.
L:SAB
L:AD/SS

OFFICE ▶	L:SAB	L:SAB	L:AD/SS			Mems
SURNAME ▶	CFerrell:bas	WPGammill	HRDenton			
DATE ▶	8/28/72	8/31/72	9/1/72			

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SAN ONOFRE 2/3

ACCIDENT CONSEQUENCES

Accident	Two-Hour Site Boundary Doses (Rem)		LPZ Course Of Accident Doses (Rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss-of-Coolant	120	5	70	2
Refueling	6	<1	<1	<1
Gas Decay Tank Rupture	0	4	0	<1

Assumptions

(1) Pasquill Type "E," $\bar{\mu}$ = 2.0 METERS/SEC

(2) Containment leak rate = 0.3%/day
For First 24 hours, 0.15%/day
For time periods greater than one day

DISTRIBUTION:
Docket (2)
PWR-3 Reading
RP Reading
OGC
RABirkel, L
CMiles, OIS
VHWilson, L (2)

Docket Nos. 50-361
and 50-362

AUG 3 1972

R. A. Miller, Office of Administration, Regulation

DISPLAY ADVERTISING FOR NOTICE OF HEARING ON APPLICATION FOR CONSTRUCTION PERMITS FOR SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

Please request display advertising of the attached notice in connection with Southern California Edison Company and San Diego Gas and Electric Company's application for construction permits for the San Onofre Nuclear Generating Station, Units 2 and 3. The notice should be placed in the following newspapers:

Sun Post
1542 El Camino Real
San Clemente, California 92672

Orange Coast Pilot
Costa Mesa, California 92626

Santa Ana Register
P. O. Drawer 1318
Santa Ana, California 92702

Los Angeles Times
Orange County Edition
1375 Sunflower Avenue
Costa Mesa, California 92626

A similar notice has been forwarded to the Office of the Federal Register for publication on or about August 11, 1972. The enclosed notice should be forwarded to the above newspapers with a request for publication by August 11, 1972.

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
Notice for Display Advertisement

FOR CONCURRENCES SEE DOCKET NO. 50-361

OFFICE ▶	AD:PWR x7415	L:PWR-3	L:PWR-3	L:AD/PWRs		Memo
SURNAME ▶	VHWilson:tls	RABirkel	KRGoller	RCDeYoung		
DATE ▶	8/2/72	8/2/72	7/2/72	8/2/72		

Form AEC-318 (Rev. 9-53) AECM 0240 U. S. GOVERNMENT PRINTING OFFICE: 1970 O - 405-346

DISTRIBUTION:
Docket (2)
RP Reading
PWR-3 Reading
OGC
RABirkel, L
VHWilson, L (2)

Docket Nos. 50-361
and 50-362

AUG 3 1972

Chief, Public Proceedings Branch
Office of the Secretary

FEDERAL REGISTER NOTICE - SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

Two signed originals of a FEDERAL REGISTER notice identified as follows are enclosed for your transmittal to the Office of the Federal Register for filing and publication:

SOUTHERN CALIFORNIA EDISON COMPANY
SAN DIEGO GAS AND ELECTRIC COMPANY

NOTICE OF HEARING ON APPLICATION FOR CONSTRUCTION PERMITS

Twelve additional conformed copies of the notice are enclosed for your use.

Please include the enclosed list of names in your Certificate of Service.

Original Signed By
K. R. Goller

Karl R. Goller, Chief
Pressurized Water Reactors
Branch No. 3
Directorate of Licensing

Enclosures:
1. Two Signed Originals
2. Twelve Conformed Copies
3. Service List

OFFICE ▶	AD:PWR x7415 VHW	L:PWR-3 RB	L:PWR-3 KRG			
SURNAME ▶	VHWilson:esp	RABirkel	KRGoller			
DATE ▶	8/2/72	8/2/72	8/2/72			

Form AEC-318 (Rev. 9-53) AECM 0240

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of SOUTHERN CALIFORNIA EDISON COMPANY SAN DIEGO GAS & ELECTRIC COMPANY (San Onofre Nuclear Generating Station, Units 2 and 3	)))))))	Docket Nos. 50-361 and 50-362

NOTICE OF HEARING ON APPLICATION FOR CONSTRUCTION PERMITS

Pursuant to the Atomic Energy Act of 1954, as amended (the Act), and the regulations in Title 10, Code of Federal Regulations, Part 50 "Licensing of Production and Utilization Facilities," and Part 2, "Rules of Practice," notice is hereby given that a hearing will be held, at a time and place to be set in the future by an Atomic Safety and Licensing Board (Board), to consider the application filed under the Act by the Southern California Edison Company and the San Diego Gas and Electric Company (the applicants), for construction permits for two pressurized water nuclear reactors designated as the San Onofre Nuclear Generating Station, Units 2 and 3 (the facilities), each of which is designed for initial operation at approximately 3390 thermal megawatts with a net electrical output of approximately 1140 megawatts. The proposed facilities are to be located at the applicants' site at Camp Pendleton, San Diego County, California.

The Board will be designated by the Atomic Energy Commission (Commission). Notice as to its membership will be published in the FEDERAL REGISTER.

The date and place of a prehearing conference and of the hearing will be set by the Board. In setting these dates due regard will be had for the the convenience and necessity of the parties or their representatives, as well as of the Board members. Notices of the dates and places of the prehearing conference and the hearing will be published in the FEDERAL REGISTER.

Upon receipt of a favorable report prepared by the Advisory Committee on Reactor Safeguards and upon completion by the Commission's regulatory staff of a favorable safety evaluation of the application and an environmental review, the Director of Regulation will consider making affirmative findings on Items 1-3, a negative finding on Item 4, and an affirmative finding on Item 5 specified below as a basis for the issuance of construction permits to the applicants.

Issues Pursuant to the Atomic Energy Act of 1954, as amended

1. Whether in accordance with the provisions of 10 CFR §50.35(a):

   (a) The applicants have described the proposed design of the facilities including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public;

   (b) Such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the final safety analysis report;

(c) Safety features or components, if any, which require research and development have been described by the applicants and the applicants have identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and

(d) On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facilities, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facilities can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

2. Whether the applicant is technically qualified to design and construct the proposed facilities;

3. Whether the applicant is financially qualified to design and construct the proposed facilities; and

4. Whether the issuance of permits for construction of the facilities will be inimical to the common defense and security or to the health and safety of the public.

<u>Issue Pursuant to National Environmental Policy Act of 1969 (NEPA)</u>

5. Whether, in accordance with the requirements of Appendix D of 10 CFR Part 50, the construction permits should be issued as proposed.

In the event that this proceeding is not a contested proceeding, as defined by 10 CFR §2.4(n) of the Commission's "Rules of Practice," the Board will (1) without conducting a de novo review of the application, consider and determine the issues of whether the application and the record of the proceeding contain sufficient information, and the review of the Commission's regulatory staff has been adequate, to support the findings proposed to be made by the Director of Regulation on Items 1-4 above, and to support, insofar as the Commission's licensing requirements under the Act are concerned, the construction permits proposed to be issued by the Director of Regulation; and (2) determine whether the environmental review conducted by the Commission's regulatory staff pursuant to Appendix D of 10 CFR Part 50 has been adequate.

In the event that this proceeding becomes a contested proceeding, the Board will decide any matters in controversy among the parties and consider and initially decide as issues in this proceeding, Items 1-5 above as a basis for determining whether the construction permits should be issued to the applicants.

With respect to the Commission's responsibilities under NEPA, and regardless of whether the proceeding is contested or uncontested, the Board will, in accordance with section A.11 of Appendix D of 10 CFR Part 50, (1) determine whether the requirements of section 102(2)(C) and (D) of NEPA and Appendix D of 10 CFR Part 50 have been complied with in this proceeding; (2) independently consider the final balance among conflicting factors contained in the record of the proceeding with a view to determining the appropriate action to be taken; and (3) determine whether the construction permits should be granted, denied, or appropriately conditioned to protect environmental values.

The application for construction permits, the applicants' Environmental Report and Supplemental Environmental Report, and, as they become available, the report of the Commission's Advisory Committee on Reactor Safeguards, the proposed construction permits, the applicants' summary of the application, the Safety Evaluation by the Commission's regulatory staff, the Commission's Draft and Final Environmental Statements, and the transcripts of the prehearing conference and of the hearing will be placed in the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., where they will be available for inspection by members of the public.

Copies of those documents will also be made available at the San Clemente Public Library, 233 Granada Street, San Clemente, California, for inspection by members of the public between the hours of 10:00 A.M. and 9:00 P.M. on Monday through Thursday, and between the hours of 10:00 A.M. and 5:00 P.M. on Friday and Saturday. Copies of the applicants' Environmental Report and Supplemental Environmental Report (to the extent of supply), and, when available, the ACRS report, the regulatory staff's Safety Evaluation and the Draft and Final Environmental Statements may be obtained by request to the United States Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing.

Any person who wishes to make an oral or written statement in this proceeding setting forth his position on the issues specified, but who does not wish to file a petition for leave to intervene, may request permission to make a limited appearance pursuant to the provisions of 10 CFR §2.715 of the Commission's "Rules of Practice." Limited appearances will be permitted at the time of the hearing at the discretion of the Board. Persons desiring to make a limited appearance are requested to inform the Secretary of the Commission, United States Atomic Energy Commission, Washington, D. C. 20545, not later than thirty (30) days from the date of publication of this notice in the FEDERAL REGISTER.

Any person whose interest may be affected by the proceeding, who does not wish to make a limited appearance and who wishes to participate as a party in the proceeding must file a petition for leave to intervene.

Petitions for leave to intervene, pursuant to the provisions of 10 CFR §2.714 of the Commission's "Rules of Practice," must be received in the Office of the Secretary of the Commission, United States Atomic Energy Commission, Washington, D. C. 20545, Attention: Chief, Public Proceedings Branch, or the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., not later than thirty (30) days from the date of publication of this notice in the FEDERAL REGISTER. The petition shall set forth the interest of the petitioner in the proceeding, how that interest may be affected by Commission action, and the contentions of the petitioner in reasonably specific detail. A petition which sets forth contentions relating only to matters outside the Commission's jurisdiction will be denied. A petition for leave to intervene which is not timely will be denied unless, in accordance with 10 CFR §2.714, the petitioner shows good cause for failure to file it on time.

A person permitted to intervene becomes a party to the proceeding and may examine and cross-examine witnesses. A person permitted to make a limited appearance does not become a party, but may state his position and raise questions which he would like to have answered to the extent that the questions are within the scope of the hearing as specified in the issues set out above. A member of the public does not have the right to participate unless he has been granted the right to intervene as a party or the right of limited appearance.

An answer to this notice, pursuant to the provisions of 10 CFR §2.705 of the Commission's "Rules of Practice," must be filed by the applicants not later than twenty (20) days from the date of publication of this notice in the FEDERAL REGISTER. Papers required to be filed in this proceeding may be filed by mail or telegram addressed to the Secretary of the Commission, United States Atomic Energy Commission, Washington, D. C. 20545, Attention: Chief, Public Proceedings Branch, or may be filed by delivery to the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C.

Pending further order of the Board, parties are required to file, pursuant to the provisions of 10 CFR §2.708 of the Commission's "Rules of Practice," an original and twenty conformed copies of each such paper with the Commission.

With respect to this proceeding, the Commission will delegate to an Atomic Safety and Licensing Appeal Board the authority and the review function which would otherwise be exercised and performed by the Commission. The Commission will establish the Appeal Board pursuant to 10 CFR §2.785 of the Commission's "Rules of Practice," and will make the delegation pursuant to subparagraph (a)(1) of that section. The Appeal Board will be composed of a chairman, an assistant chairman, Dr. John Buck, with a third member to be designated by the Commission. Notice of the Appeal Board's membership will be published in the FEDERAL REGISTER.

UNITED STATES ATOMIC ENERGY COMMISSION

W. B. McCool
Secretary of the Commission

Dated at Germantown, Maryland
this 2nd day of August 1972.

50-361/362

JUL 28 1972

W. P. Gammill, Chief, Site Analysis Branch, L
THRU: B. Grimes, Chief, Accident Analysis Branch, L

SAN ONOFRE 2 & 3 LOCA AND REFUELING ACCIDENT CALCULATIONS WITH LATEST MODELS

At. C. Ferrell's request, the Loss-of-Coolant Accident and Refueling Accident were re-run with USAECAAR using the latest models for iodine reduction. Table I gives the doses calculated. These vary from the values in the ACRS report (dated May 28, 1971) for the following reasons:

1. Loss-of-Coolant Accident

A. Thyroid, exclusion boundary dose

The latest spray parameters and assumptions result in a spray removal rate of 7.9 $hrs.^{-1}$ for inorganic iodines and a resultant dose reduction factor of 5.4 for the 0-2 hour dose. This is greater than the previously calculated DRF of 3.7.

B. Thyroid, LPZ boundary dose

The comparison of new to old dose reduction factors is:

Time Interval	New DRF	Old DRF
0 - 8 Hours	7.9	5.5
8 - 24 Hours	9.9	6.05
1 - 30 Days	9.9	9.2

C. Whole body doses

Our method of computing average energies per disintegration has changed since the ACRS report was written. The values given in that report were based on gamma decay energies only; the data in Table I are based on gamma plus beta decay energies.

2. Refueling Accident

A. Thyroid, exclusion boundary

The difference here is due to a combination of factors resulting from a change in models.

(1) 99%, not 98%, of the iodines released are assumed to remain in the pool.

(2) The filter efficiency is assumed to be 90% for inorganic and 70% for organic with 25% of the iodines above the pool assumed to be in the organic form. This is opposed to the 90% efficiency used previously.

B. Whole body doses

The gamma plus beta energies are currently used, but the fraction released from the core is reduced from 20% of the noble gases to 10%. These two factors cancel each other out.

C. Thyroid, LPZ boundary doses

This value is the same as that reported in the ACRS report, probably due to round-off.

Elinor Adensam
Accident Analysis Branch
Directorate of Licensing

cc: K. Goller
C. Ferrell
R. Birkel
bcc: E. Adensam

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AD/SS - Rdg.

OFFICE ▶	L:AAB	L:AAB	L:AAB			Memo
SURNAME ▶	EGA EADENSAM:mj	R. Zavadoski RZAVADOSKI	BGRIMES			
DATE ▶	7/26/72	7/27/72	7/27/72			

3.9.2 ASME Code Class 2 and 3 Components

All seismic Category I components, equipment and systems in ASME Code Class 2 and 3 and outside of the reactor coolant pressure boundary, will be designed, fabricated and inspected in accordance with the requirements of the applicable codes delineated in Section 3.2.2, "System Quality Group Classification."

They will be designed to sustain normal loads, anticipated transients and the Operational Basis Earthquake within the appropriate code allowable stress limits and the Design Basis Earthquake within stress limits which are comparable to those associated with the emergency operating condition category. We consider that these stress criteria provide an adequate margin of safety for Seismic Category I systems, components and equipment.

3.10 Seismic Design of Category I Instrumentation and Electrical Equipment

The reactor protective system, engineered safety feature circuits and the emergency power system are designed to meet Category I (seismic) design criteria. The seismic requirements established by the seismic system analysis will be incorporated into the equipment specifications to insure the equipment purchased or designed will meet seismic requirements equal to or in excess of the requirement for Category I (seismic) components.

The final evaluation of the topical report "Seismic Testing of Electrical and Control Equipment (WCAP-7397-L)" has not been completed at this time, however, we anticipate no major seismic component or instrumentation qualification problems. We will pursue it as a post construction permit item.

4.2.2 Reactor Vessel Internals (Mechanical Design)

For normal design loads of mechanical, hydraulic, and thermal origin, including anticipated plant transients and the operational basis earthquake, the reactor internals will be designed to the stress limit criteria of Article 4 of the ASME Boiler and Pressure Vessel Code Section III.

Under design basis accident conditions, which include the combined loads from a recirculation line break or a steam line break plus the Design Basis Earthquake, the reactor internal components will be designed to the criteria submitted in Section 14 of the PSAR. These criteria are consistent with comparable code emergency and faulted operating condition category limits and the criteria which have been accepted for all recently licensed plants. We find these criteria acceptable. The dynamic analyses of the Watts Bar Nuclear Plant reactor internals are discussed in Section 3.9.1, "Dynamic System Analysis and Testing."

5.2.1 Design Criteria, Methods and Procedures (Reactor Coolant Pressure Boundary)

The reactor coolant pressure boundary will be a Seismic Category I system designed, fabricated, and inspected in accordance with the requirements of the applicable codes delineated in Section 3.2.2, System Quality Group Classifications. The applicable codes and code editions comply with the rules of 10 CFR 50, Section 50.55a, "Codes and Standards". The stress limit criteria specified for the normal and upset operating condition categories of the applicable codes will apply for all normal loads and anticiapted transients including the Operational Basis Earthquake.

(Under the loads calculated to result from the Design Basis Accident, the Design Basis Earthquake, and the combination of these postulated events, the components of the reactor coolant pressure boundary will be designed to the applicable emergency and faulted operating condition category limits of the appropriate codes or where the appropriate codes do not provide explicit design limits for these operating condition categories, to the criteria submitted in Appendix B of the PSAR.) The plastic instability limits allowed by NB-3200 of the Code will not be employed for pumps and valves under any loading conditions. In addition, active components, i.e., pumps and valves required to operate reliably in order to perform a safety function such as safe shutdown of the reactor or mitigation of the consequences of a pipe break will be designed to deformation limits that are consistent with operational requirements. Under these restrictive deformation criteria, calculated primary stresses will be in the elastic range. We find the above stress and deformation criteria acceptable.

(In accordance with Paragraph I.701.5.4 of the ANSI B31.7 Nuclear Power Piping Code, which requires that piping shall be supported to minimize vibration and that the designer is responsible to observe that vibration is within acceptable levels, a vibration operational test program to verify that the piping and piping restraints within the RCPB have been designed to withstand dynamic effects due to valve closures, pump trips, etc. will be performed during startup and initial operating conditions. The proposed tests and the associated actions e.g., pump trips, valve actuations, etc., that are to be used in this program will be similar to those experienced during reactor operation and provide an acceptable basis for conducting the vibration operational test program.)

(The above conclusions in parentheses assumes that the verbal statements of the applicant will be adequately documented.

Consultants

The following consultant has been requested to review and evalute the applicant's proposed seismic design criteria for structures, systems and components.

Nathan M. Newmark, Consulting Engineering Services

Urbana, Illinois

(Our consultant has reviewed the Watts Bar PSAR including applicable amendments and finds the seismic design criteria documented therein to be acceptable.)

[Our consultant has not completed his final review of the applicant's submittals. We believe that no substantive issues will arise and that the consultant's report will be completed prior to the ACRS meeting for Watts Bar.]

Distribution:
Docket No. File 50-361
Docket No. File 50-362
L: RF
L: CS RF
SEB RF
L. Shao
V. Wilson (2)
R. Shewmaker

JUN 30 1972

Docket No. 50-361
and 50-362

R. C. DeYoung, Assistant Director for Pressurized Water Reactors, L

SOUTHERN CALIFORNIA EDISON COMPANY, SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3 - SUPPLEMENTAL ACRS REPORT INPUT FROM PSAR REVIEW EVALUATION

The additional PSAR material submitted by the applicant has been reviewed and evaluated by the Structural Engineering Branch of the Containment Safety Group. An evaluation of the additional information is enclosed to serve as supplemental information to the May 28, 1971, report to the ACRS. The review and evaluation are based on information provided by the applicant through Amendment No. 13, dated May 15, 1972.

The Structural Engineering Branch found that the information relative to structural aspects is adequate and that the plant can be designed and built without undue risk to the public safety.

The details of the applicant's tendon surveillance program are to be provided prior to the ACRS meeting, but it is understood that the proposed Safety Guide will be met.

Plant Name: San Onofre Units 2 and 3

Licensing Stage: PSAR - CP Review

Docket No.: 50-361/362

Responsible Branch: PWR-3, K. Goller, Chief; R. Birkel, Project Leader

Requested Completion Date: 6/12/72

Application's Response Date: ASAP prior to ACRS

Description of Response: Commitment on tendon surveillance

Review Status: ACRS Report Input Complete except for this item

R. L. Tedesco, Assistant Director
for Containment Safety
Directorate of Licensing

Enclosure:
Supplemental ACRS Report
Input-PSAR

cc: (w/o enclosure)
A. Giambusso
W. McDonald
cc: (w/enclosure)
S. Hanauer K. Goller
J. Hendrie R. Birkel

OFFICE	L:CS/SEB	L:CS/SEB	L:CS/AD	
SURNAME	RShewmaker:ewe	LShao	RTedesco per	Memo
DATE	6-28-72	6-28-72	6-30-72	

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

DISTRIBUTION:
Docket (2)
L Reading
RP Reading
PWR-3 Reading
RABirkel, L

Docket Nos. 50-361
and 50-362

JUN 27 1972

Original Signed by
G. W. Rivenbark for

R. C. DeYoung, Assistant Director for PWR's, L
THRU: K. R. Goller, Chief, PWR Branch No. 3, L

SOUTHERN CALIFORNIA EDISON COMPANY - SAN ONOFRE, UNITS 2/3
ACRS SUBCOMMITTEE MEETING - MEETING NOTES

Enclosed are the notes from the meetings held with Southern California Edison Company and the ACRS Subcommittee in Menlo Park, California. Two attendance lists are also enclosed.

Original Signed By
R. A. Birkel

Ralph A. Birkel
PWR Branch No. 3
Directorate of Licensing

Enclosures:
1. Meeting Notes
2. Attendance List - 6/19
3. Attendance List - 6/20

cc w/encls:
R. S. Boyd, L
D. J. Skovholt, L
D. F. Knuth, L
R. R. Maccary, L
R. L. Tedesco, L
H. R. Denton, L
PWR Branch Chiefs
R. W. Klecker, L
M. Rosen, L
RO (3)
V. H. Wilson, L (2)
Meeting Attendees from REG

OFFICE ▶	L:PWR-3 x7415 RAB	L:PWR-3 KR				Memo
SURNAME ▶	RABirkel:esp	KRGoller				
DATE ▶	6/27/72	6/27/72				

ENCLOSURE NO. 1

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

SAN ONOFRE, UNITS 2/3

DOCKET NOS. 50-361/362

MEETING NOTES

Summary

A meeting was held with the San Onofre 2/3 ACRS Subcommittee on June 19 and 20, 1972, at the U. S. Geological Survey - Western Center, Menlo Park, California. The purpose of the meeting was to discuss the staff's evaluation and conclusions regarding site geology/seismicity and other items resulting from the staff review of the San Onofre 2/3 CP application. An attendance list is attached for each day of the meeting. The CP review has been completed and a July meeting with the ACRS is contemplated.

Discussion

The meeting of June 19 was primarily directed to the discussion of geology and seismicity of the site. Presentations were made to the Subcommittee by the staff and our consultants summarizing the results of our evaluation of the data and information presented by the applicant and documented in Amendment Nos. 10, 11 and 12 to the PSAR. It was indicated that our conclusions did not agree with those of the applicant, however, in discussing this matter with them it was initially agreed that the San Onofre 2/3 design should meet a design criteria of .67g. The staff indicated that the draft NOAA report directed attention to "occasional peaks of 3/4g" however it had no significance in establishing structural seismic design bases and leads to misinterpretation. NOAA also stated that the 3/4g is a high frequency spike and did not contribute to the response spectra nor becomes controlling in any way. The Subcommittee agreed and recommended that suitable changes be incorporated in the report.

Presentations regarding site geology/seismicity were made by the applicant and their consultants, but at the request of the Subcommittee were limited in nature since no unresolved considerations existed between the staff and applicant. The Subcommittee did note that consideration of near-shore generated tsunamis should be considered although this aspect should not of itself delay the review in that, if shown to be necessary, suitable provisions could be made to provide suitable tsunami protection either at the seawall or to accommodate wave runout with respect to the intake structure. It should be noted that the Subcommittee as well as the staff and our consultants agreed that this matter will not necessarily be easily resolved but that the commitment by the applicant to consider and evaluate the dynamic and static effects on the site of near-shore generated tsunamis as well as wave runout conditions, is acceptable.

Regarding the Cristianitos Fault, the applicant has committed to additional trenching to better establish the tectonic character of the fault. Observers from the staff, our consultants as well as California State representatives will inspect the trench. It was concluded that whether or not the Cristianitos Fault is active or not, the offshore fault is the controlling factor. The Subcommittee found this to be acceptable.

The Subcommittee agreed that the DBE seismic response spectra being developed for the 0.67g zero period acceleration, 2% damping, is acceptable. Overall, the Subcommittee accepted the staff's conclusion for the geology/seismicity for the site.

As a consequence of the more conservative seismic design bases, various safety related structures and other features of the plant are being redesigned by the applicant. In general, buildings will be reduced in height and increased in area to present a lower profile. Separate spent fuel pools, refueling water storage tanks, and primary plant makeup water storage tanks for each unit will be provided as well as relocation of the safety injection rooms and diesel generators. These and other design changes, as well as the delay of upwards of one year in the project resulting from the geology/seismology concern has increased the project cost approximately 15-20% (current total estimate is $1 billion).

Areas that will require modification to meet the higher seismic design basis include:

a. Shear keys on the core support barrel.

b. Control rod drive assemblies (grid or snubbers required).

c. Reactor vessel supports (upward vertical restraints).

d. Steam generator supports (upper and lower pipe snubber-tube bundle may require additional support also)

e. Pressurizer - upper support against rocking motion.

f. Reactor coolant pump - will require rigid (columns restraining vertical and horizontal motion by way of pins on each end of column allowing pump to move) rather than spring hanger supports; stronger pump casing may also be required.

g. Fuel assemblies and core.

The topics discussed during the June 20 meeting primarily evolved around ECCS performance. The applicant has indicated that their core design is similar to Arkansas Unit 2 with ECCS analysis supported by Report CENPD-46. In addition, the San Onofre 2/3 design incorporates the use of high pressure (600 psi) safety injection tanks. San Onofre 2/3 will also use pre-pressurized fuel; credit, however for ECCS performance will not be taken for this design change. The current design meets the requirements of the interim ECCS criteria. Similar to Arkansas Unit 2, the applicant has agreed to provide parametric studies to support the San Onofre 2/3 design.

Additional items discussed included quality assurance, ATWS, common mode failure, loose parts monitoring, tsunami generation, Appendix I considerations, and R&D items. The only unresolved issue at this time is tornado design criteria to be applied to the station. This matter will be further discussed with the staff with anticipation of resolution prior to the July ACRS meeting. The staff report to the ACRS will be issued WB June 26, 1972.

ENCLOSURE NO. 2

ATTENDANCE LIST

SAN ONOFRE ACRS SUBCOMMITTEE MEETING

June 19, 1972

SOUTHERN CALIFORNIA EDISON COMPANY

J. B. Moore
K. P. Baskin
H. L. Potter
D. H. Johns
C. R. Kocher
W. G. Zintl
G. S. Hunt
D. G. Moon
M. L. Hill
D. E. Nunn

U. S. GEOLOGICAL SURVEY

J. G. Vedder
R. H. Morris
J. I. Ziony
H. C. Wagner
J. M. Buchanan
F. A. McKeoun

SAN DIEGO GAS & ELECTRIC COMPANY

R. G. Lacy

COMBUSTION ENGINEERING

V. C. Hall
H. B. Smith
H. V. Tease
P. L. Borkoski

NATIONAL OCEANIC & ATMOSPHERIC ADMINISTRATION

J. Devine*

---

*Denotes part-time

WOODWARD-McNEILL & ASSOCIATES

R. L. McNeill

BECHTEL CORPORATION

P. Dragolovich
L. G. Hinicelman
G. S. C. Wang
R. J. Kosiba
P. Koss

CHICKERING & GREGORY

D. R. Pigott
F. S. Bayley
M. Ott

ACRS

S. Bush
H. Isbin
W. Kaufman
H. Mangelsdorf
A. O'Kelly
B. Page, ACRS Consultant
M. White, ACRS Consultant
J. Wilson, ACRS Consultant
J. Hard, ACRS Staff

AEC - STAFF

R. A. Birkel
R. W. Klecker
R. T. Dodds
W. P. Gammill
D. Lange
A. Cardone*

---

*Denotes part-time

ENCLOSURE NO. 3

ATTENDANCE LIST

SAN ONOFRE ACRS SUBCOMMITTEE MEETING

June 20, 1972

SOUTHERN CALIFORNIA EDISON COMPANY

J. B. Moore
L. D. Hamlin
O. J. Ortega
K. P. Baskin
H. R. Ray
C. R. Kocher
D. E. Nunn
W. G. Zintl
D. F. Martin

COMBUSTION ENGINEERING

H. B. Smith
J. D. Crawford
P. L. Borkoski
W. H. Higgins
H. V. Tease
V. C. Hall
W. A. Goodwin
C. M. Berlinger

BECHTEL

G. S. C. Wang
P. Dragolovich
P. Koss
R. J. Kosiba
L. H. Curtis
R. L. Rogers
P. J. Speidel
G. H. Rohde
D. C. Unruh

SAN DIEGO GAS & ELECTRIC COMPANY

R. G. Lacey

CHICKERING & GREGORY

D. Pigott
M. Ott
F. S. Bayley

ACRS

S. Bush
H. Mangelsdorf
H. Isbin
A. O'Kelly
R. Allemann, ACRS Consultant
J. Hard, ACRS Staff

AEC - STAFF

R. A. Birkel,
R. C. DeYoung
R. W. Klecker
D. Lange
R. Dodds

Docket Nos. 50-361
and 50-362

JUN 20 1972

B. Grimes, Chief, Accident Analysis Branch, OADSS

SAN ONOFRE 2/3 REALISTIC ACCIDENT ASSESSMENT

Enclosed is the realistic accident writeup for the San Onofre draft environmental statement. We understand you will fill in the dose table and transmit the completed package to Environmental Projects Branch No. 1. A population table for the year 1980 is also enclosed as well as the plant power level and exclusion radius.

Original Signed by
K. R. Goller

Karl R. Goller, Chief
PWR Branch No. 3
Directorate of Licensing

Enclosure:
Realistic Accident Writeup

cc w/encl:
R. C. DeYoung, L
J. Youngblood,
E. G. Adensam, L
R. A. Birkel
V. H. Wilson, L (2)

DISTRIBUTION:
Docket (2)
L:Reading
PWR-3 Reading

FOR CONCURRENCES SEE DOCKET NO. 50-361

OFFICE ▶	L:PWR-3 x7415	L:PWR-3				Memo
SURNAME ▶	RABirkel:tls	KRG KRGoller				
DATE ▶	6/19/72	6/19/72				

Form AEC-318 (Rev. 9-53) AECM 0240 U. S. GOVERNMENT PRINTING OFFICE : 1970 O - 405-346

VI. ENVIRONMENTAL IMPACT OF POSTULATED ACCIDENTS

A. PLANT ACCIDENTS

A high degree of protection against the occurrence of postulated accidents at the San Onofre Nuclear Generating Station, Units 2 and 3, is provided through correct design, manufacture, and operation and the quality assurance program used to establish the necessary high integrity of the reactor system. Deviations that may occur are handled by protective systems to place and hold the plant in a safe condition. Notwithstanding this, the conservative postulate is made that serious accidents might occur, in spite of the fact that they are extremely unlikely, and engineered safety features are installed to mitigate the consequences of these postulated events. The probability of occurrence of accidents and the spectrum of their consequences to be considered from an environmental effects standpoint have been analyzed using best estimates of probabilities and realistic fission product release and transport assumptions. For site evaluation in our safety review, extremely conservative assumptions were used for a purpose of comparing postulated doses resulting from a hypothetical release of fission products from the fuel, against the 10 CFR Part 100 siting guidelines. The computed doses that would be received by the population and environment from actual accidents would be significantly less than those calculated for our site evaluation. The Commission issued guidance to applicants on September 1, 1971, requiring the consideration of a spectrum of accidents with assumptions as realistic as the state of knowledge permits. The applicant's response was contained in the "Supplement to Applicant's Environmental Report, Construction Permit Stage" dated December 22, 1971.

The applicant's report has been evaluated, using the standard accident assumptions and guidance issued as a proposed amendment to Appendix D of 10 CFR Part 50 by the Commission on December 1, 1971. Nine classes of postulated accidents and occurrences ranging in severity from trivial to very serious were identified by the Commission. In general, accidents in the high consequence end of the spectrum have a low occurrence rate, and those on the low consequence end have a higher occurrence rate. The examples selected by the applicant for these cases are shown in Table VI-1. The examples selected are reasonably homogeneous in terms of probability within each class, although (1) the release of the waste gas decay tank contents is considered as more appropriately in Class 3, and (2) the steam generator tube rupture as more appropriately in Class 5. Certain assumptions made by the applicant do not exactly agree with those in the proposed Annex to Appendix D, but the use of alternative assumptions does not significantly affect overall environmental risk.

Staff estimates of the dose which might be received by an assumed individual standing at the site boundary in the downwind direction, using the assumptions in the proposed Annex to Appendix D, are presented in Table VI-2. Staff estimates of the integrated exposure that might be delivered to the population within 50 miles of the site are also presented in Table VI-2. The man-rem estimate was based on the projected population around the site for the year 1980.

To rigorously establish a realistic annual risk, the calculated doses in Table VI-2 would have to be multiplied by estimated probabilities. The events in Classes 1 and 2 represent occurrences which are anticipated during plant operation and their consequences, which are very small, are considered within the framework of routine effluents from the plant. Except for a limited amount of fuel failures and some steam generator leakage, the events in Classes 3 through 5 are not anticipated during plant operation, but events of this type could occur sometime during the 40 year plant lifetime. Accidents in Classes 6 and 7 and small accidents in Class 8 are of similar or lower probability than accidents in Classes 3 through 5 but are still possible. The probability of occurrence of large Class 8 accidents is very small. Therefore, when the consequences indicated in Table VI-2 are weighted by probabilities, the environmental risk is very low. The postulated occurrences in Class 9 involve sequences of successive failures more severe than those required to be considered in the design bases of protective systems and engineered safety features. The consequences could be severe. However, the probability of their occurrences is so small that their environmental risk is extremely low. Defense in depth (multiple physical barriers), quality assurance for design, manufacture and operation, continued surveillance and testing, and conservative design are all applied to provide and maintain the required high degree of assurance that potential accidents in this class are, and will remain, sufficiently small in probability that the environmental risk is extremely low.

Table VI-2 indicates that the realistically estimated radiological consequences of the postulated accidents would result in exposures of an assumed individual at the site boundary to concentrations of radioactive materials within or comparable to the Maximum Permissible Concentrations (MPC) of Appendix B, Table II, 10 CFR Part 20. Table VI-2 also shows that the estimated integrated exposure of the population within 50 miles of the plant from each postulated accident would be orders of magnitude smaller than that from naturally occurring radioactivity, which corresponds to approximately 568,000 man-rems per year based on a natural background of 100 mrem/yr.

When considered with the probability of occurrence, the annual potential radiation exposure of the population from all the postulated accidents is an even smaller fraction of the exposure from natural background radiation and, in fact, is well within naturally occurring variations in the natural background. It is concluded from the results of the "realistic" analysis that the environmental risks due to postulated radiological accidents are exceedingly small.

TABLE VI-1

CLASSIFICATION OF POSTULATED ACCIDENTS

AND OCCURRENCES

Class	AEC Description	Applicant's Example(s)
1.0	Trivial Incidents	Not Considered
2.0	Small releases outside containment	Miscellaneous small spills and leaks outside containment
3.0	Radwaste system failures	Radwaste systems failures-release of 10% of Gas decay tank contents, and failure of liquid radwaste primary ion exchanger
4.0	Fission products to primary system (PWR)	Failed Fuel
5.0	Fission products to primary and secondary systems (PWR)	Failed Fuel and steam generator tube leak plus loss of load
6.0	Refueling accident	Fuel handling accident in containment
7.0	Spent fuel handling accident	Fuel handling accident-fuel handling building
8.0	Accident initiation events considered in design basis evaluation in the SAR	Gas decay tank rupture (100% contents), steam line break, steam generator tupe rupture, Control rod ejection, Loss-of-coolant pipe break.
9.0	Hypothetical sequence of failures more severe than Class 8	Not Considered

TABLE VI-2

SUMMARY OF RADIOLOGICAL CONSEQUENCES

OF POSTULATED ACCIDENTS

Class	Event	Estimated Fraction of 10 CFR Part 20 limit boundary (1)	Estimated Dose to population in 50 mile radius man-rem
1.0	Trivial Incidents	2/	2/
2.0	Small releases outside containment	2/	2/
3.0	Radwaste system Failures		
3.1	Equipment leakage or malfunction		
3.2	Release of waste storage tank content		
3.3	Release of liquid waste storage contents		
4.0	Fission products to primary system (PWR)		
4.1	Fuel cladding defects	N.A.	N.A.
4.2	Off-design transients that induce fuel failure above those expected	N.A.	N.A.
5.0	Fission products to primary and secondary systems (PWR)		
5.1	Fuel cladding defects and steam generator leaks	2/	2/
5.2	Off-design transients that induce fuel failure above those expected and steam generator leak		
5.3	Steam generator tube rupture		

- continued -

TABLE VI-2 (concluded)

6.0	Refueling accidents		
6.1	Fuel bundle drop		
6.2	Heavy object drop onto fuel in core		
7.0	Spent fuel handling accident		
7.1	Fuel assembly drop in fuel rack		
7.2	Heavy object drop onto fuel rack		
7.3	Fuel cask drop	N.A.	N.A.
8.0	Accident initiation events considered in design bases analysis report SAR		
8.1	Loss-of-Coolant Accidents Small Break Large Break		
8.1a	Break in instrument line from primary system that penetrates the containment	N.A.	N.A.
8.2a	Rod ejection accident (PWR)		
8.2b	Rod drop accident (BWR)	N.A.	N.A.
8.3a	Steam line breaks (PWR's outside containment) Small Break Large Break		

---

(1) Represents the calculated fraction of a whole body dose of 500 mrem, or the equivalent dose to an organ.

2/ These releases are expected to be in accord with proposed Appendix I for routine effluents (i.e., 5 mrem per year to an individual from all sources).

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

1980 POPULATION DATA

0-1	0
1-2	105
2-3	500
3-4	4,540
4-5	7,560
5-10	13,600
10-20	191,800
20-30	416,000
30-40	1,641,000
40-50	3,395,000
Total	5,680,000

Power level - 3410 MWt
Exclusion Radius - 800 meters (2625 ft)

man-rem
5,680,000 x .100 background Rem

= 568,000 man-rem/yr

Files

JUN 16 1972

Richard C. DeYoung, Assistant Director for PWR's, L

SAN ONOFRE NUCLEAR GENERATING STATION - UNITS 2 AND 3, DOCKET NOS. 50-361/362

The PSAR submitted by the subject applicant has been reviewed and evaluated by the Reactor Systems Branch. A final evaluation of the material as it applies to the System Quality Group Classifications and within the scope of review of this Branch is enclosed. This final evaluation supersedes that of May 26, 1971 and has been revised to reflect significant changes submitted in Amendment 13 and proposed by the applicant in recent discussions.

Original Signed By
V. A. Moore

Donald F. Knuth, Assistant Director
for Reactor Safety
Directorate of Licensing

Plant Name: San Onofre Nuclear Generating Station, Units 2 and 3
Licensing Stage: Operating License
Docket Number: 50-361/362
Branch requesting assistance: PWR 3
Project Leader: Ralph Birkel
L:TR Branches involved: Reactor Systems Branch
Description of Request: PSAR review pertaining to System Quality Group Classifications.
Review Status: Complete

cc w/o encl:
A. Giambusso, L:RP
W. G. McDonald, L:OPS
R. R. Maccary, L:RS

cc w/encl.
S. H. Hanauer, DRTA
J. Hendrie, L:TR
R. Klecker, L
K. Goller, L
R. A. Birkel, L
V. Stello, L
R. Kirkwood, L

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DATE ▶	6/15/72	6/16/72	6/16/72			

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SAN ONOFRE NUCLEAR GENERATING STATION-UNITS 2 AND 3
DOCKET NOS. 50-361/362

System Quality Group Classifications

The applicant has applied a system of code classification groups to those pressure-containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety. These classification groups Nuclear Code Groups A, B and C, and Non-nuclear Code Group D generally correspond to the Quality Group Classification System in Safety Guide 26. The codes applicable to the components in each of the applicants classification groups are identified in Tables ~~CS-1~~ and ~~CS-2~~.

We and the applicant are in agreement on the application of the code classification groups for the reactor coolant pressure boundary and other fluid systems important to safety identified in Safety Guide 26. The applicant has supplied Piping and Instrumentation Diagrams identifying the boundary limits of each classification group for the reactor coolant pressure boundary and other fluid systems important to safety.

For the Coolant Radwaste System and the Boric Acid Recycle System classified as Code Group D, the applicant will provide documentation that the failure of components in these systems would not result in calculated potential exposures in excess of 0.17 rem whole body (or its equivalent to parts of the body) at the site boundary or beyond.

We find that the system quality group classifications as specified by the applicant are acceptable.

JUN 1 6 1972

Richard C. DeYoung, Assistant Director for PWR's, Directorate of Licensing

SOUTHERN CALIFORNIA EDISON COMPANY, SAN ONOFRE UNITS 2 & 3 (CP)
DOCKET NOS. 50-361 & 50-362

The information submitted by the applicant, including Amendment No. 13, has been reviewed by the Materials Engineering Branch, L. Out sections of the Safety Evaluation are enclosed.

At a technical meeting on June 12, 1972, the applicant submitted the information relating to operating limitations and containment leakage testing verbally and agreed to provide written documentation prior to final publication of the AEC Safety Evaluation. This documentation should be obtained and reviewed prior to the ACRS meeting.

Original Signed by
R. R. Maccary

DKT # 50-361

R. R. Maccary, Assistant Director
for Reactor Safety
Directorate of Licensing

Enclosure:
Materials Engineering Branch Safety
Evaluation for San Onofre 2/3 (CP)

cc w/o encl:
A. Giambusso, L
W. G. McDonald, L

cc w/encl:
J. M. Hendrie, L
K. R. Goller, L
R. Birkel, L
S. S. Pawlicki, L
M. B. Fairtile, L
Docket File
L, Reading File
L:MTEB File

R. Gustafson, L

OFFICE	L:MTEB	L:MTEB	L:RS			Memo
SURNAME	MBF M. Fairtile:jm	SS. Pawlicki	R. R. Maccary			
DATE	6/14/72	6/15/72	6/15/72			

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SAN ONOFRE UNITS 2 & 3 (CP)

DOCKET NOS. 50-361/362

SAFETY EVALUATION - MATERIALS ENGINEERING BRANCH, L

REACTOR COOLANT SYSTEM

Fracture Toughness

For the pressure retaining components of the reactor coolant pressure boundary made of ferritic materials, materials acceptance testing was performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III (1968 Edition). Dropweight NDT data as well as Charpy V-notch energy curves have been obtained for the plates and major forgings in the reactor vessel.

To establish operating limitations during startup and shutdown of the reactor coolant system, the applicant has agreed to follow Appendix G, "Protection Against Non-Ductile Failure," of the recently revised ASME Code, Section III, fracture toughness rules (Code Case 1514). The applicant will submit specific operating limitation curves at the operating license stage.

Regarding the feasibility of annealing the reactor vessel, should it become necessary because of radiation embrittlement, the applicant has stated that the vessel could be maintained at 650°F temperature for one week by means of pump heat without major difficulties. This annealing cycle will allow partial recovery of the fracture toughness.

Reactor Vessel Material Surveillance Program

The proposed material surveillance program complies with the proposed AEC

§ 50.55a Appendix H, "Reactor Vessel Material Surveillance Program Requirements," and is consistent with programs that have been accepted for previous similar PWR plants. The program is acceptable with respect to the number of capsules, number and type of specimens, withdrawal schedule, and retention of archive material. We conclude that the proposed program will adequately monitor neutron radiation induced changes in the fracture toughness of the reactor vessel material.

Sensitized Stainless Steel

The applicant has stated that significant sensitization of all non-stabilized austenitic stainless steel within the reactor coolant pressure boundary will be avoided through materials selection and control of all welding and heat treating processes. The precautions will include control of preheat and interpass temperatures and control of heat input during the welding operations; control of weld metal composition to promote an austeno-ferrite duplex structure; checking core structure and weld procedure qualification welding by the Strauss test; and not allowing use of furnace sensitized steel for the valves, piping, and pumps.

The sensitization of reactor vessel nozzle safe-ends will be eliminated because the main coolant piping, except for the pressurizer surge pipe, is low alloy ferritic steel with stainless steel cladding on the inside, instead of the generally used stainless steel piping. Additionally, stainless steel components and piping will be joined to ferritic steel nozzles by buttering the ferritic steel with Inconel 182, prior to post-welded heat treatment, and by later shopwelding an annealed stainless steel safe-end to the Inconel 182 buildup using 182 filler metal.

We conclude that the planning to avoid sensitization of austenitic stainless steel during the fabrication period is acceptable.

Pump Flywheel Integrity

The applicant's specifications for the materials, design, fabrication and inspection procedures for the flywheels are in accordance with the AEC Safety Guide 14, Reactor Coolant Pump Flywheel Integrity, and are acceptable.

Inservice Inspection

The applicant has stated that the inservice inspection program for the reactor coolant pressure boundary will comply with Section XI of the ASME Pressure Vessel and Boiler Code. Access will be provided for each applicable component in accordance with the requirements for inspection given in Table IS-261 of Section XI.

The reactor is being designed to allow either external or internal remote inspection. Access provisions for future remote inspections are based on the most conservative size estimates of the remote equipment now being developed. Remote equipment for inservice inspection is presently under development.

We conclude that the access provisions and planning for the inservice inspection program are acceptable.

Leakage Detection System

The leakage detection system proposed for the reactor coolant pressure boundary is sensitive, includes diverse leak detection methods, and is provided with suitable control room alarms and readouts. The major components of the system are the containment atmosphere particulate and gaseous radioactivity monitors, and the level and flow indicators on the containment sump. Indirect indication of leakage can be obtained from the containment pressure, temperature and humidity indicators. We conclude that the proposed leakage detection system has the capability to detect small cracks in the reactor coolant pressure boundary.

CONTAINMENT

Leakage Testing Program

The primary reactor containment and components which will be subjected to containment test conditions will be designed so that periodic integrated leakage rate testing can be conducted at peak accident pressure.

Penetrations, including personnel and equipment hatches and airlocks, and isolation valves, are being designed with the capability of being individually leak tested at peak accident pressure.

We conclude that design of the containment system will permit the conduct of the containment leak test program in compliance with the AEC proposed "Reactor Containment Leakage Testing for Water Cooled Power Reactors," § 50.54(o), Appendix J, published in the Federal Register on August 27, 1971.

ENGINEERED SAFETY FEATURES

Inservice Inspection Program

The applicant has stated that mechanical systems outside the primary pressure boundary, such as engineered safety features, will be functionally tested at periods frequent enough to verify the continued integrity and operability of such systems. Access will be provided to the main steam and engineered safety features to perform routine and special inspections. Plans for inspections are currently under development.

We conclude that the access provisions for inservice inspection of the engineered safety features of this plant are acceptable.

JUN 16 1972

Richard C. DeYoung, Assistant Director for PWR's, Directorate of Licensing

SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3, DOCKET NOS. 50-361/362

The final evaluation for the subject plant which was prepared by the Mechanical Engineering Branch, dated May 26, 1971, has been revised to reflect significant changes submitted in Amendments through No. 13. New report sections are enclosed as direct replacements for those sections of the May 26, 1971 evaluation which have undergone major revisions. Tentative conclusions, for which confirmation is still required are enclosed in parentheses; the material in brackets provides a summary of actions to be taken to resolve issues still open at this final evaluation review stage.

DKT #50-361

Org Sgd by DFL for

R. R. Maccary, Assistant Director
for Reactor Safety
Directorate of Licensing

Enclosure:
Final Evaluation - Mechanical for
San Onofre

cc w/encl:
J. M. Hendrie, L
K. R. Goller, L
D. F. Lange, L
R. A. Birkel, L
J. P. Knight, L
N. H. Davison, L

cc w/o encl:
A. Giambusso, L
W. G. McDonald, L

OFFICE ▶	L:ME JPK	L:ME NHD	L:ME DFL	L:RS DFL for		Mcm
SURNAME ▶	J. Knight:jm	N. Davison	D. Lange	R. Maccary		
DATE ▶	5 6/15/72	6/15/72	6/15/72	6/15/72		

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FINAL EVALUATION

SAN ONOFRE NUCLEAR POWER PLANT UNITS 1 AND 2

DOCKET NOS. 50-361/362

3.6 Criteria for Protection Against Dynamic Effects Associated with a Loss-of-Coolant Accident

The applicants proposed pipe rupture criteria does not require protection of all plant features vital to safety for all postulated pipe ruptures or for all pipe break locations which may be postulated (e.g. the applicant states that limited containment liner damage could occur for some postulated main steam line ruptures.) We shall require that the applicant proceed on the basis that no steps will be taken to negate the provision of protection for all plant features vital to safety against a postulated break at any point in the RCPB or main steam piping while the question of acceptable postulated pipe break locations is being resolved on a generic basis.

3.7.1 Seismic Input

(The seismic design response spectra for the SSE and 1/2 SSE produce amplification factors of 3.5 between the period range of .2 to 1.0 sec. and of greater than 1 in the period range .2 to .033 seconds for 2% damping. All other damping values in the high frequency range are drawn parallel to the 2% damping curve starting at a period of .2 sec. The structure and equipment damping is in the range of damping factors which have been accepted for all recently licensed plants. The modified time history to be used for component equipment design is adjusted in amplitude and frequency to envelope the response spectra specified for the site. The high "g" ground motion and the associated high soil stress levels for the San Onofre site will filter the high frequencies and produce lower acceleration response levels below .2 sec.)

An OBE vibratory ground motion for continued reactor operation will not be used by the applicant. The applicant further states that he will not comply with the proposed rule "Seismic and Geologic Siting Criteria" Appendix A of 10 CFR Part 100 and that he will not require any inspection of Category II items after any earthquake exceeding .05 g. Upon final issuance of the proposed rule as a regulation, we intend to impose on the applicant the requirements of any new rules of this type at the operating license review stage.

(The above assumes that the applicant will adequately document verbal agreements with the staff.)

3.7.2 Seismic System Analyses
3.7.3 Seismic Subsystem Analyses

Modal response spectrum multi-degree-of-freedom and normal mode-time history methods will be used to develop the seismic design bases for all Category I structures, systems, and components. Governing response parameters will be combined by the square root of the sum of the squares to obtain the modal maximums when the modal response spectrum method is used. (The absolute sum of responses will be used for in-phase closely-spaced frequencies.) Floor spectra inputs to be used for design and test verification of structures, systems and components are generated from the normal mode-time history method. A vertical seismic-system dynamic analysis will be employed to account for significant vertical amplifications for the seismic design of structures, systems, and components. The resultant composite critical damping for the soil-structural system of 10% of the critical damping using a proportional damping approach will be used for the approximate nonproportional damping analysis. The applicant will also make comparisons of the nonproportional damping approach with the modal synthesis damping technique. The seismic methods and procedures that will be used for the design of structural systems and components including the Nuclear Steam Supply Systems will be clarified.

(The above assumes that adequate documentation will be provided by the applicant in Amendment 14.)

3.7.4 Seismic Instrumentation

(The applicant states that a seismic acceleration monitoring system that will automatically detect and record the seismic activity acceleration response of important features of the nuclear power plant will be engineered to ensure complete fulfillment of the AEC Safety Guide 12.)

(The above assumes that the applicant will adequately document verbal agreements with the staff.)

3.7.5 Seismic Design Control Measures

The quality assurance requirements for Category I (seismic) structures, systems, and components are stated in Amendment 4 to the application. We believe that these quality assurance provisions, which the applicant states were implemented for all items designated as seismic Category I for design, comply with the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants" of 10 CFR 50.

3.9.1 Dynamic System Analyses and Testing

The loads which would result from the postulated break of a reactor inlet pipe and the attendant stresses and deformations in the core support structures will be calculated by quasi-dynamic methods employing the peak of the axial pressure pulse during subcooled blowdown calculated from the WHAM computer code and the Seal-Shell-2 computer program to calculate stresses and deflections.

Analysis of the reactor internals response to a postulated reactor outlet pipe break will consider both the impact forces on the upper guide structure during subcooled blowdown and the dynamic response of the upper guide structure during two-phase blowdown.

We have informed the applicant of our concern that the margin of safety for reactor internals may not be sufficiently conservative when determined by other than applicable dynamic analyses for all blowdown flow regimes. The applicant and his nuclear steam system supplier, Combustion Engineering, have agreed to review the responses of the San Onofre internals under LOCA conditions using state of the art dynamic analysis techniques. Combustion Engineering expects to submit the analyses in topical report form by June 1972.

Operating dynamic loads for design will be determined on the basis of analyses performed for similar design plants. Natural frequencies calculated for the internals show that they are remote from the frequencies induced by known excitation sources such as pump blade passing and vortex shedding pressure flucturations, and forces resulting from the response of dominant coupled structures. The applicant states that recent tests at the Palisades plant have shown the analytical methods employed to be valid and that additional confirmation of the design methods employed will be submitted in the form of reports containing correlations of predictions and measurements obtained during the preoperational vibration test programs to be conducted at the Maine Yankee and Fort Calhoun plants. No internal vibration measurements are presently contemplated since the applicants present program considers San Onofre to be a non-prototype plant as defined in AEC Safety Guide 20.

We have informed the applicant that the proposed preoperational vibration test program for San Onofre will be acceptable only if the forthcoming submittals contain substantive correlation of acceptable prototype plant test data as well as documentation which establishes that reliable analytical tools have been developed to predict the dynamic response of CE reactor internal structures to normal operational flows and anticipated flow transients. In the event the forthcoming submittals do not provide acceptable prototype data from the Palisades, Maine Yankee and Fort Calhoun Plants we shall require that San Onofre provide all the necessary design features to allow the installation of all the instruments necessary to meet the prototype requirements of AEC Safety Guide 20.

(In accordance with the provisions of the ASME Section III Nuclear Component Code, which requires that piping shall be supported to minimize vibration and that the designer is responsible to observe that vibration is within acceptable levels, a vibration operational test program to verify that the piping and piping restraints within the RCPB have been designed to withstand dynamic effects due to valve closures, pump trips, etc. will be performed during startup and initial operating conditions. The proposed tests and the associated actions (pump trips, valve actuations, etc.) that are to be used in this program will be similar to those experienced during reactor operation and provide an acceptable basis for conducting the vibration operational test program.)

(The above assumes that the verbal statements of the applicant will be adequately documented in Amendment 14.)

3.9.2 ASME Code Class 2 and 3 Components

Category I (seismic) systems, components, and equipment will be designed, fabricated, and examined, as applicable, to the ASME Section III Component Code 1971 Edition.

All Category I (seismic) systems, components, and equipment outside of the reactor coolant pressure boundary were designed to sustain the Operational Basis Earthquake within the appropriate code allowable stress limits and the Design Basis Earthquake within stress limits which are comparable to those associated with the emergency operating condition category which are within the yield strength of the material for membrane stresses. We consider that the above stress criteria provide an adequate margin of safety for Category I (seismic) systems and components.

4.2.2 Reactor Internals - Mechanical Design

For normal design loads of mechanical, hydraulic, and thermal origin, including anticipated plant transients and the Operational Basis Earthquake the reactor internals will be designed to function within the stress limit criteria of Article 4, Section III of the ASME Boiler and Pressure Vessel Code.

All internal components are designated as Category I (seismic) items and will be designed to withstand loads resulting from a Design Basis Earthquake, a Loss-of-Coolant Accident and the combination of these hypothetical events. Membrane strain limits for the internals under this combined load will correspond to an elastically calculated stress of approximately two-thirds of the specified minimum tensile strength for the applicable material at temperature. The stresses calculated to result from the combined Design Basis Earthquake and Loss-of-Coolant Accident indicate that the CEA shrouds in the first row nearest the reactor vessel outlet nozzles, slightly exceed the stress at assumed failure. However, the applicant states that all CEA's would remain insertable and that the remaining CEA shrouds and all other internal components important to safety exhibit adequate margins of safety with respect to conservatively established failure criteria assuring that core cooling and reactor shutdown capability will not be impaired.

We find the criteria employed for the design of the San Onofre Unit 2 and 3 reactor internals to be acceptable.

5.2.1 Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary (RCPB) will be designed as a Category I (seismic) system to withstand the normal loads of mechanical, hydraulic, and thermal origin, including anticipated transients, and the Operational Basis Earthquake within the stress limits of the codes cited below.

The reactor pressure vessel steam generator primary and secondary sides, pressurizer and reactor coolant pumps will be designed, fabricated, and inspected to Class 1 requirements of ASME Section III Nuclear Component Code, 1971 edition.

Piping which is part of the RCPB was designed and fabricated to the requirements of the ASME Section III Nuclear Component Code, 1971 edition. The design, fabrication, and examination criteria of the codes discussed above are equal to or exceed those accepted for all recent plants of this type. We find these Codes to be acceptable for components of the reactor coolant pressure boundary.

(Under the loads resulting from the Design Basis Earthquake, the Design Basis Accident and the combination of these events, the components of the reactor coolant pressure boundary will be designed to the acceptable stress criteria of Appendix B of the PSAR. The reactor coolant pressure boundary active components will be designed to the emergency condition stress limits for the faulted condition loadings of the code. These criteria are identical to those accepted and used in the design of RCPB components for the Calvert Cliffs Plant which was the most recently reviewed plant of Combustion Engineering design.)

[We have asked the applicant for confirmation that the unacceptable N-417-11 plastic instability limits will not be employed for the design of the reactor coolant pressure boundary components. We are also awaiting documentation of the fact that compatible system dynamic analyses and stress analyses were employed for RCPB support systems and components (e.g., elastically calculated stresses based on loads developed by elastic dynamic system analyses.]

Consultants

The following consultant has been requested to review and evaluate the applicant's proposed seismic design criteria for structures, systems and components.

Nathan M. Newmark, Consulting Engineering Services
Urbana, Illinois

(Our consultant has reviewed the San Onofre PSAR including applicable amendments and finds the seismic design criteria documented therein to be acceptable.)

[Our consultant has not completed his final review of the applicant's submittals. We believe that no substantive issues will arise and that the consultant's report will be completed prior to the ACRS meeting for San Onofre.]

June 14, 1972

Docket Nos. 50-361
and 50-362

A. Giambusso, Deputy Director for Reactor Projects, L

FLASH NOTE ON SAN ONOFRE 2/3

Your May 11, 1972, note to R. Boyd, R. DeYoung and D. Skovholt directed that an assessment be made by May 22nd of the extent and categorization of matters left "unresolved" at the CP stage of the licensing process, and that I take the lead management role in the task. On May 20th I informed you that we could not meet the May 22nd completion date because of other more immediate and higher priority assignments. I also requested an indication of the urgency of the task, since I could not see a real need for the information and it did require a significant amount of effort from an already over-taxed staff. Your note of May 26th stated that "I believe it is important enough to warrant reasonably immediate attention. It should help us ultimately. Please have it done by June 19th." I assigned Ray Klecker the task of coordinating the information that was to be provided by the PWR and BWR Group Branch Chiefs. This raw information has now been provided to him. It will take about another man-week of his time to analyze the information and document it in a suitable form for your use.

In order to increase the chances for the PWR Group to submit the San Onofre 2/3 and Zion 1/2 staff reports to the ACRS for consideration at the July meeting, I have had to take unusual measures. One of these has been to assign Ray Klecker the task of writing the Supplementary Report on San Onofre 2/3, attending the ACRS Subcommittee meeting scheduled for June 19th and 20th, and providing technical support to the project leader (Ralph Birkel) who is "buried" in the Zion 1/2 review and the Offshore Nuclear Plant review. As a result of this assignment, we will not be able to meet the June 19th date you specified for completion of the "Unresolved Matters" task. I would now expect that task to be completed in July.

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If the management steps that I have taken to expedite our review of the San Onofre 2/3 application are unacceptable, please inform me as soon as possible.

Original Signed By
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

cc: J. O'Leary
E. J. Bloch
E. G. Case
K. Goller
R. W. Klecker

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RP Reading
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R. C. DeYoung

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DATE ▶	6/14/72					

JUN 13 1972

Raymond R. Maccary, Assistant Director for Reactor Safety, Directorate of Licensing

MEETING ON SAN ONOFRE, UNITS 2 & 3, CONCERNING SEISMIC SITE FACTORS AT THE UNIVERSITY OF ILLINOIS

Enclosed is a summary of the meeting held on June 8, 1972 at the University of Illinois concerning Seismic Site sensitive factors. An attendance list is enclosed.

DKT. #50-361

Org. Sgd. by DFL

David F. Lange, Chief
Mechanical Engineering Branch
Directorate of Licensing

cc w/encl:
J. M. Hendrie, L
E. G. Case, L
R. C. DeYoung, L
D. F. Lange, L
K. K. Kapur, L
A. Giambusso, L

OFFICE ▶	L:ME	L:ME	L:RS			
SURNAME ▶	Kkapur K. Kapur	DFL D. Lange	R. Maccary			
DATE ▶	6/12/72	6/12/72	6/ /72			

Enclosure 1

San Onofre, Units 2 and 3

Summary of Meeting - June 8, 1972

Summary

A meeting was held with Southern California Edison Company on June 8, 1972 at the University of Illinois to discuss the seismic response spectra, non-proportional damping and other site sensitive factors relating to San Onofre Units 2 and 3.

Discussion

The following observations and conclusions have been made:

(1) The applicant has generated a site response spectra based upon a site dependent approach which resulted in a magnification factor of 3.375 for 2% damping. It was suggested by the AEC and the consultants that a value of 3.5 should be used to be compatible with the spectra for the plants.

(2) The cut off point for reducing the magnification factor in the site response spectra generated by the applicant is 5 cycles per second. It was agreed that this may be acceptable provided the applicant gives justification for reduction in the response spectra in the high frequency region (high stresses due to the high g levels (2/3 g) for the site).

(3) The applicant has used an approximate nonproportional damping analysis incorporating very high damping values (40 and 63 percent of the critical damping for horizontal and vertical translation respectively) for soil which will result in unconservative stresses in the reactor structure and components. The staff and the consultants expressed concern that these values were not conservative, citing several technical papers that support this conclusion. It was recommended that if adequate justification were provided, the damping values selected could be such that the resultant composite critical damping for the soil-structure system may not exceed 10% of the critical using a proportional damping approach.

    It was also recommended that the applicant should continue to make comparisons of the nonproportional damping approach with the modal synthesis technique suggested by the staff.

(4) The applicant agreed to provide details of the seismic analysis techniques and models used for the analysis of reactor internals and other components such as NSSS.

Conclusion

The applicant agreed to document all the recommendations of the staff and the consultants by June 26, 1972. However, the applicant and the staff agreed that the staff would consider, as a follow on item, a submittal by the applicant to justify the use of damping values greater than 10%.

Enclosure 2

Attendance List

San Onofre Meeting - June 8, 1972

AEC

R. DeYoung
D. Lange
K. Kapur

Newmark and Hall

N. Newmark
W. Hall
A. Robinson

Southern California Edison

K. P. Baskin
H. B. Ray
D. H. Johns

Bechtel

G. S. C. Wang
R. Kosiba
T. Kohli
P. Koss

Woodward & McNeill

R. McNeill

JUN 13 1972

K. Goller, Chief, PWR Branch #3, L
THRU: H. Denton, Assistant Director for Site Safety

SUPPLEMENTAL GEOLOGY REPORT TO ACRS - SAN ONOFRE UNITS 2 & 3
DOCKET NOS. 50-361 & 50-362

A supplemental report to the ACRS on the Geology and Seismology of the San Onofre Units 2 & 3 project is attached.

/s/ HR Denton

for W. P. Gammill, Chief
Site Analysis Branch
Directorate of Licensing

Enclosure:
As stated

cc: R. DeYoung, L
A. T. Cardone, L
R. Birkel, L

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SUPPLEMENTAL REPORT NO. 2 TO THE ACRS

SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 AND 3

SUMMARY

Since our last report to the committee, AEC staff and their consultants, the U. S. Geological Survey (USGS) and the National Oceanographic and Atmospheric Administration (NOAA), have reviewed and evaluated additional information provided by the applicant regarding the geology and seismology of the San Onofre site. The written draft reports of the consultants evaluation and conclusions are appended. The conclusions of the staff and consultants can be summarized as follows:

1. Contrary to the applicants geologic interpretations, the latest geophysical data provided by Western Geophysical Co. corroborates the existence of an extensive, southeast-trending zone of deformation offshore from the San Onofre site consisting of folds that are at least locally faulted in the upper stratigraphic horizons, and continuously faulted in depth. The zone is a continuation of the onshore Newport-Inglewood Zone at the north and appears to connect with the onshore Rose Canyon Fault in the La Jolla-San Diego area at the south. The structural zone persists both offshore and onshore through this region from Santa Monica southward at least to the Mexican border. The entire zone is considered to be seismically active.

2. The NOAA consultants recommend and the staff agrees that an acceleration of 2/3g with occasional peaks of 3/4g resulting from a strong X intensity (MM) event is adequate to represent the ground motion from the maximum earthquake likely to affect this site. These accelerations could result from an earthquake occuring within a few miles from the site. Also, it must be assumed that a similar earthquake could occur at any point along this zone of deformation.

3. The seismic activity of the Cristianitos Fault and the foundation engineering aspects of the site are still under review by the staff. In any event, the offshore fault will control the determination of the safe shutdown earthquake and the seismic design accelerations for the site. The possibility of ground displacement at the plant location appears negligible, but site exploration will be monitored closely for any branch fault indications.

Offshore Geology Evaluation

Since our last report to the committee, the applicant contracted the Western Geophysical Company to carry out extensive offshore geophysical mapping, and in addition, has obtained offshore geophysical data taken by several oil companies. In our opinion, the offshore geophysical data furnished by the applicant, confirms the existence of an extensive regional zone of deformation that contiguously includes the Newport-Inglewood Zone, South Coast Offshore Fault, and the Rose Canyon Fault. We have concluded that the offshore zone of deformation is characterized by the Newport-Inglewood Zone which does not exhibit continuous faulting in the near surface rocks and sediments; but is a continuous, linear zone of deformation and faulting at depth in basement rock. The zone of deformation extends from the Santa Monica Mountains down to at least the Mexican border. Portions of this geologic structural zone exhibits historical seismicity or other evidence from which its seismic activity can be inferred: specifically, historic seismicity of the Newport-Inglewood Zone, Quaternary displacements on the Rose Canyon Fault at the south, and suggestions of Quaternary fault displacements at places in the offshore zone. Our USGS consultants have concluded that the entire zone "must be considered potentially active and capable of an earthquake whose magnitude could be commensurate with the length of the zone, and whose mode of deformation could be similar to that of the 1933 Long Beach earthquake."

Based upon the regional structural geology, and taking into account the geologic history and historic seismicity of the onshore portions of the zone of deformation, our seismological consultant (NOAA) recommends and the staff agrees that an acceleration of 2/3g with occasional peaks of 3/4g resulting from a strong X intensity (MM) event be used to represent the ground motion from the maximum earthquake likely to affect this site.

These accelerations could result from an earthquake occuring within a few miles from the site. Also, it must be assumed that a similar earthquake could occur at any point along this zone of deformation.

Cristianitos Fault Evaluation

In our last report to the Commission, we stated that absolute dating of the most recent movement on the Cristianitos Fault was not possible. Examinations carried out by the applicant to look for geomorphic or topographic evidence of recent activity indicated that the most recent movement was at least 35,000 years ago.

The staff have recently met with a geologist with the California Division of Mines & Geology who is conducting a study of the geology of the south half of the Canada Governadora Quadrangle in Orange County, California. The area under study includes a portion of the Cristianitos Fault. His findings indicated possible Holocene displacement in a trench which he assumes exposed the western trace of the Cristianitos Fault. He also interprets certain topographic maps as showing stream offsets and abrupt stream gradient change along the Cristianitos Fault. There are other explanations of his findings, such as landsliding or other non-tectonic phenomena, which need not imply movement on the fault. We and our consultants have concluded that whether the Cristianitos Fault is active or not, the offshore fault is the controlling factor in determing the zero period ground acceleration design value at the site for the safe shutdown earthquake.

The applicant has concluded that the likelihood of ground displacement within the plant site during its life time is negligible. The staff and its consultants believe that this conclusion appears supportable, provided that faults are not encountered in the excavation at the site. The staff and USGS are presently investigating the field evidence of the State Geologist and will report their findings to the committee upon completion of the investigation.

Foundation Engineering

The applicant has not provided soil testing and foundation analyses to reflect the recommended 2/3g accelerations that the plant site may experience in the event of a SSE. The staff will review the results of such testing and analyses when they are provided by the applicant and will report the results of our review to the committee.

Docket Nos. 50-361
and 50-362

JUN 9 1972

A. Giambusso, Deputy Director for Reactor Projects, L

SAN ONOFRE 2/3 - RESULTS OF MEETING WITH CONSULTANTS

In order to expedite our review of the seismic design considerations associated with the San Onofre 2/3 facility we arranged to meet with the applicant and our consultant, N. M. Newmark Associates, at the University of Illinois on June 8, 1972. As a result of that meeting we have established basic design requirements that are acceptable to the applicant and the staff, have arranged with the applicant to have the information necessary to support these requirements submitted to the staff the week of June 26, 1972, and have arranged with Dr. Newmark to have his final report submitted to the staff during the first week in July.

From a seismic design point of view these results will permit us to seek a letter from the ACRS at the July meeting, all other issues permitting. However, it should be recognized that two non-routine procedures may be involved. First, the applicant will attempt to completely revise his PSAR to reflect the recently established 2/3g value for the DBE in time for submittal during the week of June 26, 1972. In the event that time prevents the completion of the task we plan to advise the ACRS of the revised information yet to be submitted at the time of the July meeting and of our intent to require its submittal prior to issuance of our Safety Evaluation. This we believe should be acceptable since the critical information in support of the basic design procedures and methods will be available prior to the receipt of Dr. Newmark's report (this will be a requisite for issuance of his report).

The second non-routine procedure we intend to establish involves means for the systematic review of the seismic design during the post-CP period. We believe that this should be required since it is important to assure that the design properly reflects adherence to the basic design criteria for this precedent-setting seismic loading. We intend to request that this be included as a special task in the program of inspection to be conducted

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by the Directorate of Operations with the technical assistance of the PWR Groups and the Mechanical Engineering Branch of the Directorate of Licensing. In this manner we can obtain the needed assurance of acceptable design without the need to condition the license.

There is no assurance that our San Onofre 2/3 review will be completed so as to receive an ACRS letter in July. However, the seismic design issue should not now prevent this. Credit for this should go to D. Lange and to the personnel in his Mechanical Engineering Branch. The good results of the June 8, 1972, meeting were achieved only through the prompt and detailed attention given to the problems involved in the few weeks prior to the meeting.

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for PWR's
Directorate of Licensing

cc: E. G. Case, L
F. Schroeder, L
J. M. Hendrie, L
R. R. Maccary, L
D. Lange, L
M. Rosen, L
R. W. Klecker, L
K. R. Goller, L
R. A. Birkel, L

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Docket
50-361

June 8, 1972

A. Giambusso, Deputy Director for Reactor Projects, L

ECCS EVALUATIONS

The Reactor Systems Branch (RSB) in Technical Review has been assigned the responsibility for establishing the acceptability of Emergency Core Cooling Systems (ECCS) proposed and provided in the facilities for which we are reviewing applications for construction permits and operating licenses, respectively. The responsibilities of the RSB with respect to the on-going ECCS rule-making public hearing has burdened it with a workload that prohibits them from meeting all their assigned responsibilities within the desired and recommended schedules. This has been recognized and the RSB has been directed to assign its highest priority to matters attendant to the rule-making hearing. As a result the review of ECCS matters for specific applications has, in general, been delayed although attention is being given to a few cases of special importance (at the present time this includes the small-breaks analysis for Fort Calhoun and the review of the ECCS for ice-condenser plants).

To enable the staff to proceed with its licensing activities the PWR Groups proposed that, until the RSB was able to meet all its responsibilities within the established schedules, it would review the ECCS matter on specific cases to the best of its ability. We proposed that the report on ECCS that we prepared on a given application be reviewed by the RSB to assure its acceptability. We believe that this method of proceeding with the licensing process was and is practical and that the risk of establishing a position that is subsequently determined to be unacceptable on ECCS for any facility is small.

In practice the method has not worked as well as it might. The deficiencies have resulted more from differences of opinion concerning procedures for proceeding with the licensing process rather than arguments about technical issues. For example, we prepared the supplementary report on ECCS for Fort Calhoun, obtained the concurrence of the RSB with the report. Subsequently a few items of concern developed that require some additional analysis. The RSB took the stand that it could not advise the ACRS that it was satisfied with the small break analyses and that it could not support a request for an ACRS letter. We, in projects, believe that the ACRS would have been willing to complete its review of Fort Calhoun if we had presented the full story of our concerns to them, informed them of the steps we plan to take to confirm the acceptability

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of the ECCS design, and advised them that we had no intent of issuing our Safety Evaluation until we were entirely satisfied. Our position would not have resulted in any relaxation of our safety requirements and would have permitted us to proceed with the licensing process for Fort Calhoun without the need for another time-consuming meeting with the Committee. We believe that our differences of opinion to date have been of a minor nature that do, however, cause irritations and embarrassment for beyond their significance. These could be eliminated if it were clearly directed that Reactor Projects would make decisions with respect to licensing procedures after being provided with all the available facts. This issue might have been raised on the Fort Calhoun matter but time prevented its elevation to upper management. This is usually the rule rather than the exception.

We now are attempting to take two PWR plants to the ACRS in July. We are presently planning to write the report sections on the ECCS for both plants (Zion 1/2 and San Onofre 2/3). We intend to again obtain the concurrence of the RSB with the positions we take. We request that arrangements be made at and above your level to permit Reactor Projects to make decisions as to how to proceed with the licensing process if any problems of the kind that developed with Fort Calhoun arise on either of these two cases. Without some arrangement of this type I believe that the present improvised method of handling the ECCS matter is of little utility and I would recommend that we reject it completely and leave the ECCS review to the experts for completion within schedules consistent with their capabilities.

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

cc: PWR Branch Chiefs
R. W. Klecker, L
M. Rosen, L

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50-361

JUN 6 1972

H. R. Denton, Assistant Director for Site Safety, L
THRU: W. P. Gammill, Chief, Site Analysis Branch, L

SUMMARY NOTES FROM MEETING ON SAN ONOFRE UNITS 2 & 3 AT MENLO PARK, CALIFORNIA

The meeting held in Menlo Park, California on May 17, 1972 included AEC staff, ACRS staff and consultants, and USGS and NOAA representatives. A list of attendees is enclosed.

The purpose of the meeting was to discuss the USGS draft report and the oral tentative recommendations from NOAA. The ensuing discussion generally followed the items listed in the enclosed agenda. A copy of the USGS draft report is enclosed.

The Cristianitos Fault

The findings of Mr. Paul Morton, a geologist with the California Division of Mines and Geology were discussed. He has evidence which shows Holocene displacement of a soil-bedrock contact at the Western trace of the Cristianitos Fault. He also has evidence of possible stream offsets and an abrupt stream gradient change, which could be evidence of recent Cristianitos Fault movement. In addition, there is a suspicious promontory rock outcrop which intersects a stream valley where the fault trace is thought to exist. There was general agreement amongst the meeting attendees that additional trenching and investigation is desirable. I stated that we were to meet with Mr. Morton in Los Angeles to discuss the matter and to initiate a study of the field evidence. We will inform USGS of our findings.

USGS indicated that although there is no obvious displacement of the coastal terrace deposits, there is evidence of some warping, which may indicate local tectonic instability.

Mr. Ziony stated that the fault observed where the Cristianitos Fault zone intersects the shoreline has a strike of N 60^{o}E and a NW dip, which does not conform to the strike N 10^{o}W and SW dip of the main fault. This would indicate the visible fault may be a strand of the Cristianitos Fault zone.

OFFICE ▶						M. Land
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The South Coast Offshore Fault

USGS concluded that the applicant has not demonstrated that there are discontinuities offshore between the Newport-Inglewood Fault, the South Coast Offshore Fault, and the Rose Canyon Fault extension. They conclude that the faults form a continuous structure with characteristics similar to the Newport-Inglewood as observed onshore.

In his review of the offshore profiles and data, Mr. H. Wagner interpreted more faults in the upper strata than did Western Geophysical. He was able to extend the Rose Canyon Fault surface trace further north in the upper horizons than did Western, which strongly implies a connection with the South Coast Offshore Fault.

H. Wagner noted a velocity discrepancy in the offshore structural high as it was characterized by Western Geophysical, which cast doubt on the favorable interpretation placed upon the "structural high" by the applicant.

Dr. Page noted the dissimilarities between the offshore structure and the San Andreas Fault. He pointed out that the frequency of earthquakes and the amount of displacement per event is smaller on the Newport-Inglewood than on the San Andreas. Page characterized the offshore structure as a continuous zone of deformation but not necessarily having a continuous fault plane except at great depth. Dr. Wilson agreed with this characterization of the offshore structure. Apparently, all agreed that the offshore structure is not the classic San Andreas type.

Dr. Coulter indicated that with the commonly mentioned probability of 10^{-6} as a guide, it is hard to distinguish between the San Andreas and the Newport-Inglewood in terms of great earthquake potential.

Vibratory Ground Motion

Wilson recommended determining the earthquake model for the site by means of rock fracture mechanics methods. A lengthy philosophical discussion on the subjects of fault length and recurrance intervals vs. magnitude, and distance and amplitude of ground motion vs. frequency followed.

Dr. Coulter pointed out that, as a result of the Bolsa Island review, a blue-ribbon panel of experts recommended that the Newport-Inglewood fault zone be assigned a magnitude 8 for the safe shutdown earthquake. Mr. J. Devine stated that NOAA had not completed their review but that they tentatively recommend a SS earthquake of about magnitude 7 3/4 and

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Intensity XI based on the geologic model described at the meeting. This they believe would result in ground motions at the site having accelerations of 2/3g to 3/4g. Devine indicated that NOAA, so that they may fulfill their responsibility, should participate with the design engineers in defining how their "g" value recommendations should be applied. He indicated that the application of a standard response spectrum to their recommendation may not be satisfactory, as NOAA would also like to recommend the duration of strong motion to be used in the dynamic design. Steinbrugge agreed that the "g" value can be significantly modified by the amplification factors and design methods employed so that the end result can vary in conservatism.

(On May 24, 1972 NOAA gave their final recommendation. They recommend that the proposed plant be designed to a magnitude 7 1/2 safe shutdown earthquake model with an Intensity greater than X but less than XI. The horizontal "g" value recommendation was put at 2/3g with isolated peaks of 3/4g and the vertical acceleration value recommendation was 2/3 the horizontal acceleration.)

A. T. Cardone, Geologist
Site Analysis Branch
Directorate of Licensing

Enclosures:
1. List of Attendees
2. Agenda
3. USGS Draft Report

cc w/encl: J. Hendrie, L
F. Schroeder, L
R. DeYoung, L
K. Goller, L
R. Birkel, L

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SURNAME ▶	ATC ATCardone:eaj	WPG WPGammill				
DATE ▶	6/2/72	6/6/72				

MENLO PARK MEETING

Attendance List

AEC	ACRS	USGS	NOAA
Denton, H. R.	Page, B.	Coulter, H.	Devine, J.
Gammill, W. P.	Wilson, J.	Houser, F.	
Cardone, A. T.	Steinbrugge, K.	McKeown, T.	
	Hard, J.	Yerkes, R.	
		Ziony, J.	
		Vedder, J.	
		Castle, R.	
		Wagner, H.	

AGENDA

1. Discuss USGS draft report.

2. Re: The Cristianitos fault

   a) Its activity

   b) Its potential for ground displacement at the site

   c) Its potential for generating in a direct or complimentary manner earthquake vibrations at the site.

   d) The relationship of the Cristianitos to any offshore structure.

3. Re: The South Coast Offshore fault

   a) Its activity

   b) Is it a through going structure of the classic San Andreas Type?

   c) Can it generate great earthquakes?

4. Re: Vibratory Ground Motion Analysis

   a) What is the length of the structural model, and what is the assumed rupture length?

   b) What is the sense of movement? How can be preclude a significant vertical displacement component, which could be assumed to result in increased ground motion?

   c) What is the significant vertical displacement component? (This should be discussed in light of the recent San Fernando earthquake.)

   d) What are the amplification characteristics of the site?

   e) What is the basis for assigning a "g" value to the geologic model for the site?

SOUTHERN CALIFORNIA EDISON CO. AND SAN DIEGO ELECTRIC CO.
SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3,
SAN DIEGO COUNTY, CALIFORNIA

REVIEW OF RELEVANT GEOLOGY

The onshore and offshore geologic data for the proposed San Onofre nuclear generator plants--units 2 and 3--presented in the PSAR (Preliminary Safety Evaluation Report), its amendments, and its appendices have been reviewed and evaluated. The evaluation in particular has been made with respect to the significance of the data for postulating the most appropriate fault model to consider as a source of the Safe Shutdown Earthquake may effect the site. Also, included in the evaluation is consideration of possible faults within or near the site area.

This review supplements the preliminary review transmitted to Director of Division of Reactor Standards, AEC on July 12, 1971 by the Chief Geologist, U. S. Geological Survey V. E. McKelvey, in the sense that many explicit details in the preliminary review are not repeated here. The present review does summarize the pertinent data from all amendments and treats in appropriate detail the large amount of offshore geophysical data acquired since the preliminary review.

In addition to the data in the PSAR, appropriate literature was reviewed and the knowledge and interpretations of geologists and geophysicists most familiar with the geology of the San Onofre area was solicited and used.

A field inspection of the site and of outcrops cut by the nearby Cristianitos fault zone was made in company with the applicant's geologists on February 8, 1972.

The principal objectives of this review are (1) to assess the applicability of the fault model used by the applicant for establishing the Safe Shutdown Earthquake, (2) to ascertain whether or not the applicant has established that no active faults of any size occur within the area; and (3) to ascertain whether or not the applicant has provided adequate data to establish the tectonic stability of the area.

With regard to the first problem, the USGS stated in the preliminary review of July 12, 1971 that "On the basis of present data there appears to be a linear zone of deformation in Tertiary and Quaternary rocks and sediments that extends southeastward from Santa Monica to at least the Mexican border." Reference is also made to the review of the geology of the site of San Onofre Nuclear Generating Station #1 transmitted by the Director, U. S. Geological Survey to AEC Director of Regulations dated October 11, 1966, in which it was stated: "The submarine extension of the Newport Inglewood zone, as mapped by Emery Energy (1960 fig. 68) also trends northwestward and is about 7 miles southwest of the site." The additional geophysical data obtained by Western Geophysical Company tends to corroborate this statement despite the applicant's claims to the contrary. The zone of deformation is at least 240 km in length, and is approximately 5 miles offshore in the vicinity of the plant site.

The available data relevant to the second problem does not show evidence of any faults that cut the rocks underlying the planned reactor facilities.

With regard to the third problem apparent warping of the terrace on which the site is located indicates that the site was or is in an area that has been or is tectonically unstable.

The San Onofre reactor site area for unit 1 and the proposed units 2 and 3 is adjacent to a beach and on thick-bedded, poorly cemented sandstone and minor thin beds of gravel and siltstone of the Plio-Pleistocene San Mateo Formation. The beds dip 10-15° to the northwest. Drilling near the site shows the formation to be more than 900 feet thick; mapping indicates the formation may be approximately 2,000 feet thick. The San Mateo formation is unconformably overlain by marine terrace deposits and unconformably underlain by siltstone, sandstone, tuff, and diatomite of the Capistrano Formation of late Miocene age. Samples collected by USGS from correlative terrace deposits about 4 1/2 miles northwest of the site were dated at 70,000-130,000 years BP according to the applicant (Amendment 1, p. 2. 9-3b).

The Cristianitos fault, a generally northerly trending fault, about three-fourths of a mile southeast of the site, is the major structural feature nearest the site. Exposure of parts of this fault at the coast and at the Plano Trabuco excavations made by the applicant shows that the overlying terrace deposits are not offset by the fault. As the fault zone is several hundred feet wide, however, and its full width is not exposed anywhere, some question remains about the age of the most recent movements on the fault.

Whether or not displacement of terrace deposits has occurred on the Cristianitos fault, evidence of warping of these deposits both locally and regionally is suggested by work of B. J. Szabo and J. G. Vedder (1971). They show deformation of the first terrace between Laguna Beach and San Clemente. The applicant correctly assumes (Amendment 1, p. 2.9-3b and Amendment 6, p.2.9-5) that the terrace deposits overlying the Cristianitos fault at the coast are associated with the same terrace as cited above. Warping of the terrace must have occurred, however, to account for approximately 55 feet difference in shore angle elevation of the terrace between the site and Dana Point, about 10 miles northwest of the site. Intermittent, if not continuous local tectonic instability, therefore, may be assumed during the last 100,000 ± 30,000 years.

The regional setting of the San Onofre site is important with respect to the location of major active fault zones and to evaluating how an earthquake in one of the zones may effect the site. According to the applicant (Amendment 6, p. 2.9-7) the "plant site is approximately 60 miles from the San Andreas fault, 45 miles from the San Jacinto fault, 23 miles from the Whittier-Elsinore fault and 18 miles from the southeast terminus of the Newport-Englewood zone." The proximity of the Newport-Inglewood zone and its alignment with the South Coast (SCO) Offshore fault, which has been recognized to various degrees in most seismic profile data (Amendment 11, appendix 2E-9) is the major structural problem to be considered. Initial offshore shallow penetration sparker data indicated discontinuous faulting and folding in the SCO fault zone. The latest investigations used a deep penetration seismic reflection method. These investigations incorporated the interpretation of 1,000 miles of deep reflection data, 450 miles of magnetic-profiling data, and 7 seismic refraction lines. In addition, the applicants and their consultants reviewed published gravity data, unpublished proprietary data obtained by others, and approximately 275 miles of United States Geological Survey (USGS) sparker reflection profiling (Amendment 11, appendix 2E, p. 2E-2). The applicant's principal conclusions as the result of this work are as follows (Amendment II, appendix 2E, p. 2E-17):

A. Displacement on the fault, even at the deepest level of investigation, dies out to the northwest near Laguna Beach and to the southeast near Oceanside.

B. In Miocene-age rocks of Horizon B, the fault shows much less displacement than in older rocks, and its trace appears as a series of short, discontinuous breaks.

C. The fault is not connected with other faults at its extremities.

D. Investigation with side-scan sonar (refer to appendix 2A, and figure 8), high-resolution profiling (refer to appendix 2C), and Western's fathometer data as seen in figure 2E-4A show no evidence of tectonic displacement on this fault since the sea floor was eroded some 18,000 years ago during an earlier glacial age (appendix 2A). No evidence has been found in these investigations of surface rupture or offset of geomorphic features, or to suggest post-Miocene or Quaternary activity on the fault.

E. There is no macroseismicity associated with the fault (refer to appendix 2A, figure 3).

(To continue with the quotation)

Based on the foregoing, it is considered that significant activity has not occurred in the South Coast Offshore Fault in post-Miocene time. This fault must not have generated a large earthquake since Upper Miocene time; i.e., millions of years ago." (Appendix 2E, p. 2E-17 and 2E-18).

The U. S. Geological Survey does not accept these conclusions for the principal reasons are outlined below:

Much of the interpretation upon which the conclusions are based is dependent upon the validity of the offshore structure contour maps. An adequate explanation of the precision of determination of the B structural horizon is lacking. Least reliable is that part of the Horizon B map south of the San Clemente core hole, but for stratigraphic reasons. Although the applicant assumes Horizon B is in miocene rocks, no Miocene rocks are known onshore between Oceanside and San Diego, and their presence offshore cannot definitely be demonstrated. Contours for that part of the Horizon B map may represent a marker stratigraphically lower than Upper Miocene, possible even within the Eocene. This concept might account for the apparent continuity on the Horizon B map of the offshore Rose Canyon fault, in contrast with the supposed discontinuous nature of the South Coast Offshore fault at an allegedly equivalent stratigraphic level. Another indication of the unreliability of the B horizon is that the Cristianitos fault is shown by the sparker profiles and on the C horizon map but not on the B horizon map. Many of the higher resolution seismic profiles show more faulting and more recent faulting than acknowledged by the applicants reports. Furthermore, Western Geophysical Company states that no continuous reflecting horizon is present in the Miocene section across the entire area (Amendment 11, appendix A-1, p.29). Also, only faulting defined by vertical offsets is considered, without assurance that no lateral displacements have taken place on some, ohterwise, unidentified faults. Interpretations of the existence of all faults significant and the ages of any particular faults on the basis of the Horizon 'B' map, therefore, are not considered complete nor accurate.

Horizon C is also unreliable. According to Western Geophysical Company "...Horizon C constitutes the base of sediments with formation interval velocities less than 12,000 feet per second,..." (Amendment 11, appendix A-1, p. 26). As an example of the unreliability, USGS analysis of velocity data in the "offshore high" indicates a maximum velocity there of about 6,200 feet per second.

A

Connections between the Newport-Inglewood zone, South Coast Offshore fault, and the Rose Canyon fault cannot be precluded even if the Horizon B and C maps were to be assumed correct. The possible junction of the Newport-

Inglewood fault zone and the South Coast Offshore fault is in an area of no data at the Horizon C level and is marked on the Horizon B map either as "no shallow resolution" or "possible shallow fault."

The assertion of the applicants that the Rose Canyon fault turns inland near Oceanside cannot be supported. True, there are a number of north- and northeast-striking faults along the coast between La Jolla and Oceanside, but these are lapped by Pleistocene Lindavista Formation; in contrast, the onshore Rose Canyon fault is characterized by vertical separations of the Lindavista Formation greater than 200 feet. A connection between the offshore Rose Canyon and the South Coast Offshore fault is probable in the area shown as "possible intrusive." This is indicated on USGS profiles that show sea floor displacement on an extension of the Rose Canyon fault to and beyond the alleged intrusives. A

[illegible]

Termination of the Newport-Inglewood zone by the San Joaquin Hills offshore structural high, as inferred by the applicant (Amendment 11, appendix 2E, p. 2E-18) is not consistent with the geology of the landward extension of the high. The applicant's interpretation is largely based on what he believes is an apparent thin onlap of upper Miocene rocks on the high, which suggests the high has been stable since middle Miocene time. Actually the folding of the onshore continuation (applicant's structural correlation) of the high--San Joaquin anticline--is known to be late Pliocene-late Pleistocene in age. No evidence is presented to demonstrate that the age of the rocks and the age of deformation in the offshore high is any different than the onshore ~~(Amendment 11, A1, p. 41). "The South Coast Offshore fault, appropriately 40 miles in length~~ rocks. Additionally, Western Geophy reports "offset of the anticlinal axis occurred after formation of the San Joaquin anticline," making SCO movement young indeed alleged

The contrasts in tectonic style between the Los Angeles basin and the Peninsular Range Province (Amendment 11, appendix E, p. 2E-16) are not clearly defensible. Probably the most prominent structural feature[5] on figure 9 are the northwest-trending faults, subparallel to and aligned with the Newport-Inglewood zone. This zone apparently has a very prominent offshore counterpart in the South Coast Offshore fault. Comparison of section L-M with section C-D clearly shows prominent similarities in structure between the basin and province.

According to Western Geophysical Company interpretations "The South Coast Offshore fault, approximately 40 miles in length, was active after the Offshore San Joaquin structure was formed." (Amendment 11, appendix A-1, p. 41). As the onshore extension of this structure is known to be late Pliocene-late Pleistocene in age, even the deeper parts of the SCO fault observed on the aquapulse reflection profiles must be post-Pleistocene. The applicant has not demonstrated conclusively whether or not Holocene or later movement can be positively identified on this fault. ~~but~~ However, such ocean bottom evidence is not necessarily germane to postulating that it is an active zone. The evidence of movement on it is likely to be a function of the mode and depth of deformation. It is reasonable to assume that the deformation has been similar to that which has occurred in the Newport-Inglewood fault zone. As pointed out by the applicant's Board of Review concerning the Newport-Inglewood zone "...(1) no member of the fault complex is known to cut strata younger than late Pleistocene, and (2) no surface-ground displacement is known to have accompanied historic earthquakes associated with the zone." (Amendment 6, appendix 2C, p. 13).

The lack of macroseismicity is not necessarily an indication that the SCO fault system is wholly inactive. It is well known that parts of many major active fault zones, such as ~~for example,~~ the San Andreas, may exhibit ~~have parts with~~ little or no seismicity.

The data presented by the applicant lead us to the conclusion that the Newport-Inglewood zone of folds and faults, the South Coast Offshore fault, and the Rose Canyon fault zone cannot be disassociated. Instead, an extensive, linear zone of deformation, at least 240 km long, extending from the Santa Monica Mountains to at least Baja, California, seems well established by the present evidence. This was the tentative assessment offered also by E. H. Baltz in the preliminary draft, transmitted on July 2, 1971, by ~~V. E. McKelvey,~~ the Chief Geologist, to E. G. Case, Director, Division Reactor Standards. This assessment is confirmed by the new information, and is still pertinent ~~seems applicable still~~ ~~especially with regard to inadequacy of fault determination in or near the sea floor. It is quoted as follows:~~

"A southeast-trending offshore extension of the Newport-Inglewood "fault" or zone has been mapped in previously published reports as passing offshore of San Onofre and extending to positions shown, variously, as near La Jolla and south of San Diego. (For example see: M. L. Hill, 1971, fig. 1; King, 1968; Allen and others, 1965, pl. 1; and Emery, 1960, fig. 68.) The offshore geophysical data recently obtained by the applicants and by the USGS appear to corroborate the published maps that indicate an extensive linear offshore zone of deformation although there are uncertainties owing to gaps in the data between Encinitas and La Jolla."

It is concluded, therefore, that Deformation on the ~~NI fault~~ Newport-Inglewood zone has gone on intermittently or continuously since at least middle Miocene and that ~~there is~~ no evidence has been presented to show that the stress system is inactive or altered. As a number of earthquakes have occurred near the north end of this zone in historic time (the largest is the 1933 Long Beach M 6.3 earthquake) and the south end ~~has~~ shows evidence of movement in Quaternary time, the whole zone must be considered potentially active and capable of an earthquake whose magnitude ~~may~~ could be commensurate with the length of the zone, and whose mode of deformation could be similar to that of the 1933 Long Beach earthquake.

"... The present offshore data on the zone of deformation are not adequate to determine precisely whether some individual faults are continuous for distances of many miles in the rocks and sediments at and near the sea floor, or whether they are discontinuous as in the shallow part of the onshore Newport-Inglewood segment. ..."

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Original Signed By
K. R. Goller

R. C. DeYoung, Assistant Director for PWR's, L
THRU: K. R. Goller, Chief, PWR Branch No. 3, L

SOUTHERN CALIFORNIA EDISON COMPANY - SAN ONOFRE UNITS 2/3
MEETING SUMMARY OF MAY 31, 1972

Enclosed are the notes from the meeting held with Southern California Edison Company on May 31, 1972, in Bethesda, Maryland. An attendance list is also enclosed.

Original Signed By
R. A. Birkel

Ralph A. Birkel
PWR Branch No. 3
Directorate of Licensing

Enclosures:
1. Meeting Notes
2. Attendance List

cc w/encls:
A. Giambusso, L
R. S. Boyd, L
D. J. Skovholt, L
H. R. Denton, L
R. L. Tedesco, L
E. G. Case, L
R. R. Maccary, L
D. Muller, L
D. F. Knuth, L
PWR Branch Chiefs
R. W. Klecker, L
CS Branch Chiefs
RS Branch Chiefs
SS Branch Chiefs
RO (3)
Meeting Attendees from REG
V. H. Wilson, L (2)

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SURNAME ▶	RABirkel	KRGoller				
DATE ▶	6/2/72	6/4/72				

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ENCLOSURE NO. 1

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

SAN ONOFRE UNITS 2/3

DOCKET NOS. 50-361/362

MEETING NOTES

Summary

A meeting was held with the applicant on May 31, 1972, in Bethesda, Maryland to discuss the conclusions of the geological and seismological evaluation of San Onofre Units 2 and 3 by the staff and its consultants, USGS and NOAA. Based upon information and data provided by the applicant and extensive review and discussion with both the applicant and their consultants, the staff concluded that a horizontal ground acceleration value of 2/3g with peaks of 3/4g should be used for the Design Basis Earthquake for San Onofre Units 2 and 3.

Discussion

Subsequent to meetings held in the spring and summer of 1971 between the staff and SCE, regarding Units 2 and 3 geology and seismology, the applicant informed us in September, 1971, that they were developing a program involving additional extensive offshore investigations and explorations to provide additional data and information to serve as a basis for the structural geology to be used to establish the seismological characteristics of the site. In April, 1972, SCE filed two amendments (Nos. 11 and 12) providing this additional information. The staff and our consultants, USGS and NOAA have reviewed and evaluated this information during the course of many technical meetings and pursuant to this evaluation the staff has concluded that a horizontal ground acceleration value of 2/3g with peaks of 3/4g should be used for the Design Basis Earthquake for San Onofre Units 2 and 3. The applicant was informed of this conclusion and was complemented by the staff for the extent of their effort in obtaining the detailed geological data and information provided in the application.

A brief elaboration of the bases for the staff conclusions was presented by the staff including comments by our consultants. USGS indicated that evidence is not conclusive that basement break is discontinuous and cannot be disassociated with onshore structures. NOAA indicated that a strike-slip structure was basically assumed although some vertical and horizontal movement was included; however, the structure is not similar to the San Andreas fault.

SCE expressed appreciation for the time and effort expended by the staff and consultants in the review and evaluation performed. SCE accepted the 2/3g DBE in light of non-proportional damping and other site sensitive factors that could be employed in the design of the station. The applicant agreed that development of suitable response spectra for the higher g value is required and would be provided as soon as possible. It was also agreed that the site related non-proportional damping concept should be further reviewed with the staff and our seismic consultant, N. Newmark. Subsequently, a meeting to discuss this aspect was scheduled with N. Newmark and the applicant for June 8, 1972.

The staff did hasten to point out that the onshore Christianitos fault matter was still open pending further evaluation of the findings by the California Bureau of Mines & Geology and that it may not be completely resolved until excavation at the site is performed. SCE agreed with these comments.

The applicant expressly requested staff consideration in meeting with the ACRS in their June meeting to discuss the staff geology/seismology conclusions. The staff agreed that there would be merit in this approach and indicated that every effort would be taken to schedule the meeting including preparation and issuance to the ACRS of our staff report on geology and seismology in the remaining 7 days prior to the June ACRS meeting. The staff was informed by the ACRS subsequent to the meeting that due to the unavailability of the ACRS consultants on such short notice, a June meeting could not be accommodated.

ENCLOSURE NO. 2

ATTENDANCE LIST

MAY 31, 1972, MEETING

SOUTHERN CALIFORNIA EDISON COMPANY

W. Gould
J. Moore
O. Ortega
K. Baskin
P. West

SAN DIEGO GAS & ELECTRIC COMPANY

M. Engler
R. Lacey

AEC STAFF

E. G. Case
R. C. DeYoung
S. H. Hanauer
H. R. Denton
W. P. Gammill
D. Lange
R. A. Birkel
A. T. Cardone

CONSULTANTS

NOAA - J. Devine
USGS - H. Coulter

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket Nos. 50-361
and 50-362

JUN 5 1972

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WITHHOLDING OF INFORMATION PURSUANT TO SECTION 2.790

By letter dated March 30, 1972, Mr. Jack B. Moore, Southern California Edison Company, transmitted proprietary data prepared by Western Geophysical Company of America on migrated depth-sections, Line WS70-3 and WS70-18, and requested that this data be withheld from public disclosure pursuant to 10 CFR 2.790(b). The request is made for the reason that this data contains information which is customarily held in confidence by Western Geophysical Company of America, its originator, and which, if disclosed would adversely affect Western Geophysical Company of America in the conduct of its business.

There is sufficient nonproprietary information provided in Amendment 11 to advise an interested member of the public on the geologic and seismic aspects of this application, but the applicant has submitted the additional proprietary data for use by the AEC in its review. Copies of this proprietary data are being made available to the AEC's consultants, the U. S. Geological Survey and the National Oceanic and Atmospheric Administration.

In view of the foregoing, I have determined that disclosure of the above data is not required in the public interest nor by 10 CFR Part 9, that disclosure would adversely affect the interests of Southern California Edison Company and Western Geophysical Company of America, and that it should be withheld from public inspection pursuant to 2.790(b) of 10 CFR Part 2.

A Giambusso
A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket Nos. 50-361
and 50-362

JUN 2 1972

A. Giambusso, Deputy Director for Reactor Projects
THRU: K. R. Goller, Chief, PWR Branch No. 3, L

FORTHCOMING MEETING WITH SOUTHERN CALIFORNIA EDISON COMPANY
SAN ONOFRE UNITS 2 AND 3

Time and Date: 10:30 A. M. - June 8, 1972

Location: N. M. Newmark & Associates
University of Illinois
Champaign, Illinois

Purpose: To discuss seismic response spectra, non-proportional damping and other site sensitive factors relating to San Onofre Units 2 and 3.

Participants: SOUTHERN CALIFORNIA EDISON COMPANY
(K. Baskin, H. Ray, P. West, et al)
AEC - Staff
(R. C. DeYoung, D. Lange, R. A. Birkel, K. Kapur)

Ralph A. Birkel
PWR Branch No. 3
Directorate of Licensing

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R. Tedesco, L
H. R. Denton, L
CS Branch Chiefs
RS Branch Chiefs
SS Branch Chiefs
PWR Branch Chiefs
R. W. Klecker, L
R. F. Fraley, ACRS
RO (4)
V. H. Wilson, L (2)
Meeting Attendees from REG

Memo

June 2, 1972

Docket Nos. 50-363
50-361
and 50-362

R. Minogue, Assistant Director for Safety and Materials Protection Standards, RS

FORKED RIVER AND SAN ONOFRE 2/3 REVIEWS

The Forked River 1 and San Onofre 2/3 construction permit applications were submitted shortly after the reorganization of the regulatory staff in early 1970. The Combustion Engineering, Inc. nuclear units for the two plants are of substantially identical design. In accord with management's description of the intent of that reorganization, we in the PWR Group of DRL proposed that we accept the responsibility for review of the instrumentation, control and electrical power aspects of the plant designs in order to free the DRS Electrical Systems Branch (Voss Moore's Branch) of a small portion of their case workload so that some part of their time could be devoted to the development of guides and standards. We based our proposal on the fact that one of the senior engineers previously reporting to Voss Moore had, in the reorganization, been assigned to the PWR Group. We contended that with his (Olan Parr's) guidance and active participation the project leaders on the two cases would be able to assure that the designs complied with the then current requirements established by Voss Moore's Branch.

Unfortunately, the reviews of both applications were interrupted for significant periods of time because of special problems that arose in other unrelated areas. Because of the elapsed time we felt that we could no longer assure that our review of the instrumentation, control and electrical areas would be consistent with current requirements without the active assistance of Voss Moore's Branch. We arranged to have Don Sullivan of that Branch assist us in completing the reviews of the applicable areas for both plants. The reviews are nearly completed; both applications are scheduled to be reviewed by the ACRS by August.

As a result of the latest reorganization, Don Sullivan is now assigned to your group. I have been informed by the Forked River 1 project leader that Don Sullivan told him that he had been directed by his Branch Chief to cease all Forked River 1 review activities. I am well aware of the difficult transitions that must be made upon a reorganization and of the urgent need to release reassigned personnel of their former responsibilities as rapidly as possible

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in order that they can meet the new responsibilities of their present assignments. However, it has always been a guiding principle in any of our reorganizations that individuals would complete previously assigned tasks if those tasks were near completion and if reassignment of the task would be inefficient to the reorganized staff. In our view, the review of the instrumentation, control and electrical areas for the Forked River 1 and San Onofre 2/3 applications can best and most efficiently be completed by Don Sullivan. For this reason, we request that Don Sullivan be permitted to complete those reviews. A prompt decision is needed on this matter since it will influence arrangements of meetings to be held next week.

Original Signed By
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

cc: A. Giambusso
E. G. Case
W. M. Morrison
V. Moore
D. Sullivan
D. Knuth
K. Goller
R. Birkel
C. Moon

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Miriam R. Evans
Director

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50-363

March 6, 1972

Mr. Frank W. Malone
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Malone:

Thank you very much for sending us the valuable documents on Forked River Nuclear Station, Docket No. 50-363, which I am sure the public find very useful.

We would appreciate it if you could send us the appropriate binders to file the amendments to these documents.

Sincerely yours,

Elizabeth H. Booth

Elizabeth H. Booth
Reference Librarian

(Mrs. R. E. Booth)

request filled
3-9-72.

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Docket Nos. 50-361
and 50-362

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A. Giambusso, Deputy Director for Reactor Projects, L
THRU: K. R. Goller, Chief, PWR Branch No. 3, L KRG

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

Time and Date: 8:30 A. M. - May 31, 1972

Location: Room P-422 - Bethesda, Maryland

Purpose: To discuss geological and seismological evaluation of the San Onofre site.

Participants: SOUTHERN CALIFORNIA EDISON COMPANY
(W. Gould, J. Moore, et al)
AEC - STAFF
(E. Bloch, E. Case, A. Giambusso, R. DeYoung, H. Denton, et al)
CONSULTANTS
(USGS and NOAA)

Ralph A. Birkel

Ralph A. Birkel
PWR Branch No. 3
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V. H. Wilson, L

RESCHEDULED DATE

Memo

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket Nos. 50-361
and 50-362

MAY 23 1972

A. Giambusso, Deputy Director for Reactor Projects, L
THRU: K. R. Goller, Chief, PWR Branch No. 3, L KRG

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

Time and Date: 9:00 A. M. - May 30, 1972

Location: Room P-422 - Bethesda, Maryland

Purpose: To discuss geological and seismological evaluation of the San Onofre site.

Participants: SOUTHERN CALIFORNIA EDISON COMPANY
(W. Gould, J. Moore, et al)
AEC - STAFF
(E. Bloch, E. Case, A. Giambusso, R. DeYoung, H. Denton, et al)
CONSULTANTS
(USGS and NOAA)

Ralph A Birkel
Ralph A. Birkel
PWR Branch No. 3
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Mem

MAY 4 1972

Docket Nos. 50-361
50-362

W. P. Gammill, Chief, Site Analysis Branch, L

GEOLOGY QUESTIONS ON SAN ONOFRE UNITS 2 & 3, DOCKET NOS. 50-361 AND 50-362

A copy of my questions on Geology which were formally submitted in a memo to Mr. K. Goller on May 2, 1972 was sent to Mr. F. Houser our USGS consultant on May 2, 1972 for his information.

A. T. Cardone
Geologist
Site Analysis Branch
Directorate of Licensing

cc: H. Denton, L

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Form AEC-318 (Rev. 9-53) AECM 0240 U. S. GOVERNMENT PRINTING OFFICE : 1970 O - 405-346

MAY 4 1972

Docket Nos. 50-361
50-362

W. P. Gammill, Chief, Site Safety Branch, L

COMMENTS AND QUESTIONS RE SAN ONOFRE UNITS 2 AND 3, DOCKET NOS. 50-361 AND 50-362

The attachment entitled, "San Onofre 2 & 3 Amendment #11 Review Comments and Questions" is the unedited comments and questions which I discussed with Southern California Edison personnel in the meeting in Bethesda held on April 17, 1972, which you attended.

A. T. Cardone
Geologist
Site Safety Branch
Directorate of Licensing

Attachment:
San Onofre 2 & 3 Amendment #11
Review Comments and Questions

cc w/encl: H. Denton, L
R. DeYoung, L
K. Goller, L

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SAN ONOFRE 2 & 3
AMENDMENT #11 REVIEW COMMENTS & QUESTIONS

INTRODUCTION:

The principal areas of concern geologically and seismologically in our evaluation of the San Onofre site have been and still are:

1. The Cristianitos fault
   a) Its potential for ground displacement at the site
   b) Its potential for generating in a direct or complimentary manner earthquake vibrations at the site.
   c) What is the relationship of the Cristianitos to any offshore structure?

2. The offshore structure
   a) If it is a fault, is it active?
   b) Is it a continuous fault and is it long enough to generate great earthquakes?
   c) Is it a through going structure of the classic San Andreas Type?

3. Finally, what ground motion resulting from the structural geologic model should be assigned at the site?
   a) What is the length of the structural model, and what is the assumed rupture length?
   b) What is the sense of movement? How can we preclude a significant vertical displacement component, which could be assumed to result in increased ground motion?
   c) What is the significant vertical displacement component? (This should be discussed in light of the recent San Fernando earthquake.)

d) What is the basis for assigning a "g" value to the geologic model for the site?

GENERAL STATEMENT:

During the early stages of the San Onofre review, the applicant presented a picture describing the intermittent faulting offshore as being shallow, discontinuous, associated with folding, anticlines, etc., and that the faulting did not extend into the basement. Now the picture has completely changed. The applicant has made further investigations and now concludes that the basement faulting found underlying the near surface faulting is not expressed near the surface, and is old, discontinuous, and segmented.

This has not been fully substantiated and further explanation will be required to justify classifying the faulting off San Onofre as inactive and incapable of creating great earthquakes.

COMMENTS & QUESTIONS RE: CRISTIANITOS FAULT

ITEM 1:

During the early stages of the San Onofre review, the applicant described the Cristianitos as inactive on the basis of:

a) The seashore fault exposure showing no displacement of the overlying terraces.

b) The trenching done at Plano Trabuco which apparently showed no displacement of overlying alluvium.

c) Small (90') fault displacement at the seashore.

d) No connection with offshore faulting.

e) No multiple movement in the past 500,000 years.

The recent data and information have apparently contradicted 4 of the above five support items.

a) Still valid

b) A California Division of Mines and Geology geologist has indicated to me that he had the fault trenched and found evidence of displaced slope wash. He believes the fault has moved during Holocene time.

c) Aquapulse information contradicts this - Displacement probably more like 900'.

d) Aquapulse information contradicts this - Connection shown in Figure 2E-2.

e) Displacement much greater than 90' could indicate multiple movement.

ITEM 2:

I have recently (April 11, 1972) spoken to Mr. Paul Morton, a geologist with the California Division of Mines and Geology, regarding a report that he is writing on an area of California which includes the north end of the Cristianitos fault. The pertinent points that came out of that conversation were:

a) Mr. Morton had the Cristianitos fault trenched and he found indications of displacement of slope wash by the Cristianitos.

b) He has observed anomalous stream gradient changes and evidence of sag ponds.

c) His conclusion is that the Cristianitos has moved during Holocene time, that is in the area of his investigation.

The applicant has been informed of Mr. Morton's findings and interpretations and has indicated he will evaluate them. He will provide AEC with the results of his evaluation.

The applicant's present position is that they trenched the fault at Plano Trabuco and found evidence of inactivity for at least 32,000 years. However, I would point out that there may be more than one strand or trace of the Cristianitos in that area and the other trace could have moved more recently.

ITEM 3:

On p. 2E-9 the applicant states that the Aquapulse data confirm that the Cristianitos fault extends seaward with decreasing displacement, terminating at the South Coast Offshore fault.

This contradicts an earlier interpretation made by the applicant that the displacement on the fault near the seashore is only 90' and that the fault terminates a short distance offshore.

Based on the data presented it would appear that the aquapulse data confirm that the displacement on the Cristianitos fault at the shoreline is significantly greater than 90'. Reasoning: if the displacement at the Plano Trabuco trench location is 5,000', as we were told, and the distance from Plano Trabuco to the shoreline is 16 miles (if we characterize the Cristianitos fault as a scissors fault), this results in a linear relationship of a 300' change in displacement per mile of fault to result in a 90' displacement at the shoreline as the applicant states. If this were so, the fault should terminate in a very short distance offshore; however, it is still going strong where it joins the South Coast Offshore fault. Conclusion: The displacement at the shoreline is far greater than 90' and may indicate multiple movements on the Cristianitos in the past 500,000 years.

The applicant has been requested to discuss the apparent offsetting of the Cristianitos fault by the South Coast Offshore fault as can be seen in Figure 2E-2.

QUESTION RE: OFFSHORE AREA

ITEM 1:

What kind of accuracy should be ascribed to the geophysical methods of determining fault displacements in the B and C horizons: The applicant has been requested to document their response.

Since, as the applicant states on p. 2E-5 in Amendment #11, in the central area offshore from San Onofre (I presume this is the offshore San Joaquin structural high) "the acoustic basement consists of San Onofre Breccia which is a poorly stratified sedimentary formation of Mid-Miocene age," how can one hope to convincingly observe the presence or absence of faulting in this matter, especially (see profile) if the faulting is strike-slip?

On p. 2.8-17 of Amendment #11, item A states that the submarine faults on the broad continental borderland are predominantly strike-slip. Since the formations and horizons are essentially flat-lying, how then can it be shown decisively be means of geophysical methods that faulting does or does not exist?

ITEM 2:

The tectonic significance of the structural high described by the applicant is not clear. He states that it segments the offshore area into 3 provinces. However, he does not explain why the 3 provinces concept should preclude a continuous offshore fault zone. The applicant should:

a) Explain why the "largest observed displacement along the South Coast Offshore fault occurs on the flank of this 'high'." (See lines 125, 127, 129, and statement on p. 31.)

b) Discuss the probability that the South Coast Offshore fault could be shown to have a 7000' lateral displacement on the basis of lateral offset between the axis of the offshore San Joaquin Hills structure and the axis of the San Joaquin Hills anticline (personnel communication reference on p. 40 of Western report).

c) Discuss the implecations of such a displacement.

COMMENTS AND QUESTIONS RE: WESTERN'S REPORT TO SCE:

ITEM 1:

Is it possible that the sparker and boomer survey techniques used by Marine Advisors, Inc., and the USGS oceanographers are more revealing at shallow depths than is the Aquapulse technique? And don't the Marine Advisors and

USGS seismic profile records show significantly more continuity of faulting offshore along the coast, where the South Coast Offshore fault would intersect the upper strata?

There is reason to doubt the interpretation by Western in that they state, "The seismic data also suggest that the Cristianitos fault extends seaward and dies out into the South Coast Offshore fault on Horizon C. In confirmation of the onshore data, it has been inactive for a long time," for it does not extend upward very far into the section and does not cut Horizon B.

The fact is that onshore the Cristianitos fault cut thru the section well above Horizon B to the top of the San Mateo formation which is thought to be Plio-pleistocene, whereas Horizon B is upper Miocene.

Further, the applicant and Western Geophysical make frequent reference to maximum displacements of a few hundred feet at the "b" horizon (Upper Miocene) on the South Coast Offshore fault, however, the displacement far up-section in the San Mateo Plio-pleistocene formation is at least in this range.

<u>ITEM 2:</u>

The Western report states:

> "Unlike the South Coast Offshore faults, the Rose Canyon fault system cuts Horizon B over its entire length offshore, It appears to turn inland near Oceanside and is separated from the South Coast Offshore fault by a series of intrusives and a series of north-south faults, which are probably extensions of the north-south fault system opposite on shore."

Why must the existence of the intrusives preclude a connection between the South Coast Offshore fault and the Rose Canyon fault?

The Western report states:

> "The Newport-Inglewood zone of deformation terminates at the Offshore San Joaquin Hills structrual high."

Yet, no significant support is given for the statment and the above quote is about all the discussion provided by Western concerning the Newport-Inglewood fault.

The Western report states:

> "The Rose Canyon fault system has been outlined from the seismic reflection data, and appears to project into the coast near Oceanside."

The location where it intersects the coast is not given.

Also by Western:

> "Because of primary concern about tectonic stability at the San Onofre plant site, the limits of the South Coast Offshore fault have been defined. It crosses the Central Region and the northwest portion of the Southeast Province about five milte offshore, and strikes northwest-woutheast. It dies out to the northwest as it approaches the Newport-Ingelwood zone of deformation."

As shown on the structure contour map of Horizon C, it also dies out to the southeast as it approahces the Rose Canyon fault. What do profile lines W8, Ws-70-26, 137 and 141 show? (Note these lines probably don't go shoreward far enough to be definitive.)

Figure 2E-2 strongly indicates that these faults are joined by the South Coast Offshore fault.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket Nos. 50-361
and 50-362

MAY 8 1972

A. Giambusso, Deputy Director for Reactor Projects, DL
THRU: K. R. Goller, Chief, PWR Branch No. 3, DL KRG

FORTHCOMING MEETING WITH U. S. GEOLOGICAL SURVEY REGARDING SAN ONOFRE 2/3 GEOLOGY

Time and Date: 8:30 A. M. - Wednesday, May 17, 1972

Location: U. S. Geological Survey - Western Center
345 Middlefield Road
Menlo Park, California

Purpose: Discuss the draft USGS and NOAA reports on geology and seismicity respectively, of the San Onofre 2/3 site.

Participants: AEC STAFF
(H. Denton, W. Gammill, A. Cardone, R. Birkel)
ACRS
(Staff - J. Hard; Consultants - J. Wilson, B. Page, C. Allen, M. White)

Ralph A. Birkel

Ralph A. Birkel
PWR Branch No. 3
Directorate of Licensing

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Meeting Attendees from REG
V. H. Wilson, DRL

Memo

Docket Nos. 50-361
and 50-362

MAY 8 1972

A. Giambusso, Deputy Director for Reactor Projects, DL
THRU: K. R. Goller, Chief, PWR Branch No. 3, DL KRG

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PWR Branch No. 3
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Meeting Attendees from REG
V. H. Wilson, DRL

Docket Nos. 50-361
50-362

MAY 2 1972

K. Goller, Chief, PWR Branch #3, DL
THRU: W. P. Gammill, Chief, Site Safety Branch, DL

QUESTIONS ON GEOLOGY AND FOUNDATION ENGINEERING - SAN ONOFRE UNITS 2 & 3, DOCKET NOS. 50-361 AND 50-362

Attached is the edited version of the questions that were informally submitted to the applicant on April 18, 1972. The applicant should be provided a copy of these questions to avoid possible misinterpretations or misunderstandings of our informal communications.

A. T. Cardone
Engineering Geologist
Site Safety Branch
Directorate of Licensing

Enclosure:
As Stated

cc: H. Denton, DL (w/enclosure)
R. DeYoung, DL (w/enclosure)
R. Birkel, DL (w/enclosure)

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DATE ▶	5/1/72	5/ /72	5/ /72			

SAN ONOFRE 2 & 3

AEC QUESTIONS - MEETING RE GEOLOGY

APRIL 17, 1972

1. Cristianitos Fault

   a. The South Coast Offshore fault offsets the Cristianitos fault and hence is younger than the Cristianitos. Since the Cristianitos has a minimum age of 35,000 to 100,000 years, should not the South Coast Offshore fault be considered as less in age and, therefore, considered active? Or, if the Cristianitos is connected with the South Coast Offshore fault, should they not be considered genetically related?

   b. Verify the 90 feet of vertical displacement of the Cristianitos fault at the coastline. It would appear to be much more as a result of the latest information in Amendment #11; thus leading to a possible conclusion that there has been more than one movement in the past 500,000 years. Explain.

   c. Paul Morton of the California Division of Mines and Geology has, by oral communication in early April with the staff, indicated his belief that there has been Holocene displacement of the Cristianitos in the Trabuco Plains area. Explain. Also what is its effect on seismic and potential surface faulting at San Onofre?

d. The Cristianitos fault is shown to extend offshore in the Western data, Horizon "C". Why is it not present in higher horizons, particularly if Horizon "B" is Upper-Miocene and Cristianitos faulting took place during post-Upper Miocene time offsetting the Pliocene San Mateo formation?

2. What is the vertical resolution in the Aquapulse data in Horizons "C" and "B"?

3. Could there be fault offset across the offshore N-S structural high in the poorly stratified San Onofre breccia that cannot be discerned from the geophysical data? What would be the significance of such offset along the trend of the South Coast Offshore fault? Explain the large offset in Horizon "B" on the South Coast Offshore fault just south of the San Joaquin High.

4. Discuss the probability that the South Coast Offshore fault could be shown to have a 7000' lateral displacement on the basis of lateral offset between the axis of the offshore San Joaquin Hills structure and the axis of the San Joaquin anticline (personal communication referenced on p. 40 Western report).

5. Make comparisons of the San Andreas fault, San Fernando (Sylmar) faults, and the South Coast Offshore fault, and discuss the following items with regard to the South Coast Offshore fault:

a) What is the length of the structural model? What is the assumed rupture length?

b) What is the sense of movement? How can we preclude a significant vertical displacement component, which could result in increased ground motion?

c) What is a significant vertical displacement component? (This should be discussed in light of the recent San Fernando earthquake.)

d) What is the basis for assigning a "g" value to the geologic model for the site?

e) Discuss the amplification characteristics of the sedimentary deposits, assuming a reasonable basement rock acceleration.

6. Provide plan and profile drawings showing the locations of all Class 1 structures, pipelines, cut slopes, etc., and show the relative location of all borings.

7. Show that in the event of failure the embankments around the plant cannot endanger any safety related structures, or provide assurance that the possibility of failure is negligible.

8. If the South Coast Offshore fault is strike slip, would the fault show up as a continuous break in the "B" Horizon and Sparker Horizons? Could this apparent decrease in displacement upward be related to changing stress conditions? (i.e., from early east-west to later north-south compression.)

9. The tectonic significance of the structural high is not clear. The applicant states that it segments the offshore area into 3 provinces. However, the applicant does not explain why the 3 provinces concept should preclude a continuous offshore fault zone.

50-361

Docket Nos. 50-361
and 50-362

APR 28 1972

Original Signed by
K. R. Goller

R. C. DeYoung, Assistant Director for PWR's, DRL
THRU: K. R. Goller, Chief, PWR Branch No. 3, DRL

SOUTHERN CALIFORNIA EDISON COMPANY - SAN ONOFRE UNITS 2 & 3
MEETING SUMMARY

Summary

A meeting was held in Bethesda, Maryland on April 17 and 18, 1972, with the Southern California Edison Company (SCE). The purpose of the meeting was to discuss the outstanding review items that are listed in abbreviated form in the enclosure (Enclosure No. 1), discuss methods of direct application of element damping and soil characteristics of the San Onofre site and briefly review and discuss site geology.

A list of attendees is also enclosed (Enclosure No. 2).

Discussion

In November, 1971, the staff developed a list of review items which would require documentation or resolution prior to completion of our review. This list of pending review items was discussed with the application during subsequent meetings and appears in abbreviated form in Enclosure No. 1. The staff discussed these and related items with SCE who agreed to address these items in Amendment No. 13.

The applicant provided a discussion of the technique of nonproportional damping as a practical method to apply appropriate damping to different elements or materials within a structural model. In particular it was indicated that the method offers the possibility of assigning high damping to the soil, moderate damping to the structural model and low damping to certain equipment such as nuclear components. In conjunction with this discussion the applicant also provided the results of effort spent in the development of elastic and damping properties of the soils and rocks at the San Onofre site. Subsequent to extensive discussion with

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DATE ▶						

APR 28 1972

the applicant, the staff indicated that the use of the non-proportional damping technique would be evaluated and considered for the San Onofre application.

During a summary type discussion of site geology, both onshore and offshore, it became evident that specific items of interest concerning geological interpretations should be elaborated upon in the PSAR. The applicant agreed to provide this elaboration in the forthcoming amendment.

At the conclusion of the meeting, SCE stated that all remaining information required to complete the application would be submitted by May 15, 1972 (Amendment No. 13). SCE again requested a staff decision concerning site geology at an early date.

Original Signed By
R. A. Birkel

Ralph A. Birkel
PWR Branch No. 3
Division of Reactor Licensing

Enclosures:
1. Pending Review Items
2. Attendance List

cc w/encls:
P. A. Morris, DRL
F. Schroeder, DRL
T. R. Wilson, DRL
R. S. Boyd, DRL
D. J. Skovholt, DRL
H. R. Denton, DRL
R. Tedesco, DRL
E. G. Case, DRS
R. R. Maccary, DRS
D. F. Knuth, DRS
PWR Branch Chiefs
R. W. Klecker, DRL
RPS Branch Chiefs
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V. H. Wilson, DRL (2)
Meeting Attendees from REG

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SURNAME ▶	RABirkel:esp	KRGoller				Mems
DATE ▶	4/28/72	4/28/72				

ENCLOSURE NO. 1

SAN ONOFRE 2/3

PENDING REVIEW ITEMS

(EXCLUSIVE OF GEOLOGY/SEISMOLOGY)

1. Design Basis Tornado

2. Reactor Coolant Pressure Boundary Faulted Stress Limits

3.a CVCS Let-down Line Code Group Classification

3.b Code Group D Criteria

4. Spent Fuel Handling System

5. Automatic Protection against Core Power Maldistributions

6. Compliance with General Design Criteria

7. Compliance with Safety Guides 1 through 23

8. ECCS Report

9. Consequences of a Single Rod Withdrawal Accident

10.a Main Steam Line Flow Restricters

10.b Main Steam Line Whipping and Containment Liner Damage

11. Reactor Vessel Internals Vibration Monitoring Program

12. Compliance with Proposed Appendices G and H of 10 CFR 50

13. Post Accident Hydrogen Gas Control System

14.a Compliance with Flywheel Integrity Safety Guide

b Requirement for Under-frequency and Under-voltage Reactor Coolant Pump Trip

15. Status of Dynamic Analysis of Reactor Vessel Internals

16. Reactor Cavity Pressure During Loss of Coolant Accident

17. Compliance with Draft Ultimate Heat Sink Safety Guide

18. Compliance with Appendix I 10 CFR 50

19. Reactor Coolant System Over-pressure Protection

20. Adequacy of Station Radiological Emergency Plan to Protect State Park Visitors

21. Compliance with Safety Guide Concerning Industrial Sabotage

22. Corrosion of the Reactor Coolant System Pressure Boundary by Leaking Boric Acid

23. Reactor Coolant Water Chemistry Limits

24. Steam Generator Tube Wall Thinning

25. Effects of Turbine-Generator Trip with Loss of all AC Power on Accidents Analyzed

26. Effect of Three Pump Operation on the Accidents Analyzed

27. Failed Fuel Detection System Sensitivity

28. Provision of Reference Plant Technical Specifications

29. Conformance with IEEE Code 344

30. Residual Heat Removal System Requirements

31. Safety Injection Tank Valve Control

32. Compliance with Proposed Appendix J 10 CFR 50

33. Adequacy of Post Accident Monitoring Provisions

34. Verification that Personnel are not Required to Leave the Control Room During a Loss of Coolant Accident

35. Discuss Compliance with IEEE Standard 344-1971 Seismic Qualification of Class I Electrical Equipment

36. IEEE Standard 338 Trial Use Criteria for Periodic Testing of Protection System

37. IEEE Standard 336 Installation Inspection and Testing Requirements for Instrumentation and Electrical Equipment During Construction

38. IEEE Standard 279-1971 Criteria for Protection Systems

39. IEEE Standard 317 dated April 1971 Standard for Electrical Penetration Assemblies in Containment Structures

40. IEEE Standard 323 Trial Use Standard General Guide for Qualifying Class I Electrical Equipment for Nuclear Power Generating Stations and IEEE Standard 334-1971 IEEE Trial Use Guide for Type Tests of Continuous Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations

41. Control-room Air Conditioning

42. Method for Periodic Testing of Engineered Safety Feature Instrumentation and Control Equipment (IEEE 279) + Safety Guide No. 22

43. Provide Information Identifying Readouts and Indications Available to the Operator for Monitoring Conditions in the Reactor Coolant System and the Containment Throughout All Operating Conditions

44. Compliance with Safety Guides 6 and 9

45. IEEE Standard 308 Criteria for Class IE Electric Systems in Nuclear Power Generating Stations

46. Fuel Oil Transfer System

47. Redundant Station Batteries and Battery Room Ventilation Systems

48. Cable Trays

ENCLOSURE NO. 2

ATTENDANCE LIST

SAN ONOFRE 2/3 MEETING

APRIL 17 AND 18, 1972

SOUTHERN CALIFORNIA EDISON COMPANY

K. Baskin 1, 2/
H. Ray 1/
P. West 1/*
O. Ortega 1/*
G. Hunt 1/*

BECHTEL

L. Curtis 1, 2/
R. Kosiba 2/
R. McChesney 2/
T. Kohi 2/

WOODWARD-McNEILL ASSOCIATES

R. McNeill 2/ - Consultant

AEC - DRL/DRS/CO

R. Birkel 1, 2/
H. Denton 1/*
K. Goller 1/*
W. Gammill 1/*
C. Ferrell 1/*
A. Cardone 2/*
J. Knight 1/*
R. Kirkwood 1/*
D. Lange 1, 2/*
M. Dunenfeld 1/*
M. Fairtile 1/*
R. Shewmaker 1/*
K. Kapur 2/
S. Hou 2/
L. Beratan 2/*

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* - Part-time
1/ - April 17, 1972
2/ - April 18, 1972

APR 12 1972

B. K. Grimes, Chief, Accident Analysis Branch, DRL
V. Benaroya, Chief, Effluent Treatment Systems Branch, DRL

DRAFT SECTIONS ON SOURCE TERM AND ACCIDENT ANALYSES FOR ES

Please provide me with your current schedule for completion, and what information you require from the Project Leader and by when on the drafts of the sections on Source Term or Accident Analyses for the DREP Environmental Statements on the following applications:

Zion 1/2
Hutchinson Island-1
Millstone-2
San Onofre 2/3
Forked River
Arkansas-2
Summer

Original Signed By
K. R. Goller

Karl R. Goller, Chief
PWR Branch No. 3
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DATE ▶	4/ /72					

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

April 6, 1972

Peter A. Morris, Director, Division of Reactor Licensing
THRU: K. R. Goller, Chief, PWR Branch No. 3, DRL KRG

FORTHCOMING MEETING WITH SOUTHERN CALIFORNIA EDISON COMPANY - SAN ONOFRE UNITS 2/3 - DOCKET NOS. 50-361/362

Time and Date: 9:00 A.M. - April 18, 1972

Location: Room P-114 - Bethesda, Maryland

Purpose: Discussion of methods of direct application of element damping and soil characteristics San Onofre site.

Participants: SOUTHERN CALIFORNIA EDISON COMPANY
(K. Baskin, H. Ray, et al)

AEC - DRL/DRS
(R. Birkel, D. Lange, K. Wichman, J. Brammer, A. Cardone)

Ralph A. Birkel
Ralph A. Birkel
PWR Branch No. 3
Division of Reactor Licensing

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J. L. Sullivan, DR
R. W. Klecker, DRL
V. H. Wilson, DRL
Proposed Principal Attendees REG

Memo

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MAR 8 1972

50-361/362

L. M. Muntzing, Director of Regulation
THRU: E. J. Bloch, Deputy Director of Regulation for Reactor Licensing

SAN ONOFRE SITE GEOLOGY AND SEISMOLOGY

Subsequent to meetings held in the spring and summer of 1971 between the staff and the Southern California Edison Company (SCE) regarding geology and seismology considerations for the proposed San Onofre Nuclear Generating Station Units 2 and 3, SCE informed us in September 1971 that they were developing a program involving additional investigation and exploration to provide additional data and information that could be used to establish appropriate seismic criteria for the San Onofre Station. Since that time they have kept us and our consultants informed of the progress of their efforts in various meetings and consultations held with them at their request. During these meetings, the most recent of which occurred on February 29, 1972, it has become increasingly evident that the information that SCE intends to submit formally will probably not significantly change our consultants' earlier views concerning the geology of the San Onofre site region. This means that we would still conclude that a continuous zone of deformation capable of generating a major earthquake exists offshore.

SCE plans to file its additional information with the Commission about April 1, 1972. Our consultants will require approximately two months to review it and then, if we follow our normal procedures with the ACRS and the Commission, it will require several additional weeks before we would be in a position to notify the applicant of our formal findings. An adverse finding on San Onofre also could have serious impact on other potential nuclear plant sites in southern California.

I suggest that we inform the Commission at this time of our consultants' adverse reactions to the information that has been presented to them to date and of the highly probable outcome of the extensive and costly effort being expended on this subject by the applicant and the staff.

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

cc: R. C. DeYoung
H. Denton
K. Goller
R. Birkel

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DATE ▶	3/8/72	3/8/72	3/ /72			

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K. Kniel
D. Vassallo
A. Schwencer
V. Wilson (2)
M. Groff, DRL

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MAR 8 1972

H. R. Denton, Assistant Director for Site and Radiological Safety, DRL
THRU: W. P. Gammill, Chief, Site Safety Branch, DRL

SAN ONOFRE UNITS 2 and 3 DOCKET NOS. 50-361 and 50-362

Attached is a summary report, written by Mr. A. T. Cardone, of the San Onofre Units 2 and 3 meeting held in Houston, Texas on February 10, 1972.

D. E. Nunn
Chief Earth Scientist
Site Safety Branch
Division of Reactor Licensing

cc: P. A. Morris
E. Case
R. DeYoung
R. Minogue
K. Goller
A. Cardone
R. Birkel

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SURNAME ▶	ATC ATCARDONE:mj	DEN DENUNN	WG WPGAMMILL			
DATE ▶	3/ 7 /72	3/ 7 /72	3/ 8 /72			

SUMMARY REPORT OF THE SAN ONOFRE UNITS 2 AND 3 MEETING IN HOUSTON, TEXAS - DOCKET NOS. 50-361 AND 50-362

On February 10, 1972 a technical meeting was held in the offices of Western Geophysical Company, a geophysical consultant to the San Onofre applicant, in Houston, Texas. The meeting included representatives from Western Geophysical, Southern California Edison, DRS, DRL, ACRS and their consultants, and DRS consultants from USGS and NOAA. A list of the attendees is attached.

The purpose of the meeting was to hear a discussion by the applicant and its consultants of the results of a recent offshore geophysical exploration program performed for or furnished to the applicant by Western Geophysical. The objective of the exploration program was to describe the offshore structural geology at depth and to develop an offshore geologic model for the site. Briefly, the model, as the applicant presented it, consists of three separate geologic areas or elements in the offshore region between Los Angeles and San Diego. The central area, which lies offshore and to the northwest of the San Onofre site, was described as structurally stable and does not have potential for tectonic movement.

Following the presentation by the applicant, AEC and ACRS staff and their consultants caucused to discuss the new geologic model presented. In the course of our discussion it became apparent that some of the consultants felt that the applicant had not provided support for the geologic model given at the meeting. The staff consultants were told that the regulatory staff felt that the geophysical data and the applicant's forthcoming amendment to the PSAR, which should contain the information presented at the meeting with modifications that reflect the comments and suggestions that we were about to make to the applicant and should contain the full geologic picture and seismic evaluation, should be reviewed before reaching any conclusions. Conclusions now would be premature.

The following comments, suggestions, and requests were made to the applicant when the meeting reconvened:

1. Provide a complete statement of the geologic and geophysical position. Correlate the subsurface offshore geology with the onshore geology to the northeast from the Los Angeles Basin down to San Diego, and use that correlation to aid in the interpretation of the offshore area.

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2. The bases for interpretations for both the gravity and magnetics map presentations should be clearly stated, and correlations to onshore gravity and magnetics should be made.

3. The 350 miles of offshore profiles made by SCE should be used as backup for the maps, and a contour map based on that 350 miles of profiles should be attempted.

4. The information on the four drill holes that were used for control should be included in the amendment, with lithologic descriptions, geophysical logs, and, specifically, the sonic velocity.

5. A tie between horizon B and the sparker surveys should be illustrated and discussed thoroughly, and the structure as displayed in both deep and shallow seismic findings of the onshore geology should be correlated.

6. On map A those sea-bottom irregularities which coincide with known faults or possible faults in the deeper seismic surveys erring on the more conservative side should be shown.

7. Provide composite maps that would include all faults shown on maps previously submitted, and indicating in some appropriate manner which horizons the faults are defined by.

8. Provide and discuss the velocity calculations used in the interpretations.

9. Develop a complete seismological model and an interpretation of the earthquake generating capacity of these models; describe and discuss the earthquake generating fault mechanism, taking into account such things as: the anticipated total length of fault displacement, and the attitude of fault movement. Determine the Safe Shutdown Earthquake (SSE), the Operating Basis Earthquake (OBE), and the corresponding seismic design accelerations. (The applicant has elected to specify on OBE.)

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In addition to the above items the applicant was asked to analyze the tsunami generating potential of the applicant's new structural geology model, and take possible tsunami effect at the site to the extent that it may alter the tsunami evaluation basis developed for Unit 1.

A. Thomas Cardone
Site Safety Branch
Division of Reactor Licensing

CRESS OFFICE ▶	DRL					
T-1001 R5-7 SURNAME ▶	ATC ATCardone:cls					
3/2/72 DATE ▶	3/ /72					

WESTERN GEOPHYSICAL

February 10, 1972

Name	Organization
David G. Moore	Southern Cal. Edison
I-Chi Hsu	Western
Byron D. Ruppel	USGS
Aart de Jong	Western
Jack B. Moore	Southern Cal. Edison
James T. Wilson	ACRS - Consultant
Ben M. Page	ACRS - Consultant
James E. Hard	ACRS - Staff
James F. Devine	NOAA
Philip J. West	Southern Cal. Edison
Gail S. Hunt	Southern Cal. Edison
Charles R. Kocher	Southern Cal. Edison
Harold P. Ray	Southern Cal. Edison
Kenneth P. Baskin	Southern Cal. Edison
A. T. Cardone	AEC - Staff
Joseph I. Ziony	USGS
R. F. Yerkes	USGS
Holly C. Wagner	USGS
F. A. McKeown	USGS
Ralph A. Birkel	USAEC - DRL
Robert Minogue	USAEC - DRS
F. N. Houser	USGS
W. C. Browder	Western
E. J. Mateke, Jr.	Western
Carl H. Savit	Western

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FEB 25 1972

P. A. Morris, Director, Division of Reactor Licensing

FORTHCOMING MEETING WITH SOUTHERN CALIFORNIA EDISON COMPANY
SAN ONOFRE UNITS 2 AND 3 - DOCKET NOS. 50-361 AND 50-362

Time and Date: 9:00 A.M. - February 29, 1972

Location: Room P-422 - Bethesda, Maryland

Purpose: Mr. Bill Gould, Senior Vice President, Southern California Edison Company has requested an opportunity to present to senior AEC management the results and conclusions of recent additional offshore exploration and investigation effecting the seismic design of San Onofre Units 2 and 3.

Participants: Southern California Edison Company
(W. Gould, J. Moore, O. Ortega, B. Laverty, P. West)
AEC
(Senior AEC management with appropriate staff members)

/S/

Harold R. Denton, Assistant Director
Site and Radiological Safety
Division of Reactor Licensing

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D. J. Skovholt, DRL
R. Tedesco, DRL
E. G. Case, DRS
L. Rogers, REP
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V. H. Wilson, DRL (2)

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SURNAME ▶	RABirkel:esp	KRG KRGoller	RCDeYoung	HRDenton		
DATE ▶	2/24/72	2/24/72	2/24/72	2/ /72		

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R. A. Birkel, PWR Branch #3, Division of Reactor Licensing

SAN ONOFRE UNIT NOS. 2 AND 3 - DOCKET NOS. 50-361 & 50-362

As you, R. Minogue, and I agreed, the following item should be communicated to Southern California Edison as an addition to those items given at the February 10, 1972, meeting with Southern California Edison in Houston, Texas.

Item: After developing a complete geologic model from the integrated on-shore and off-shore geologic information available, develop a complete seismological model and an interpretation of the earthquake generating capacity of these models; describe and discuss the earthquake generating fault mechanism, taking into account such things as: the anticipated total length of fault displacement, and the attitude of fault movement. Determine the Safe Shutdown Earthquake (SSE), the Operating Basis Earthquake (OBE), and the corresponding seismic design accelerations.

Anthony T. Cardone

Anthony T. Cardone
Special Projects Branch
Division of Reactor Standards

OFFICE ▶	DRS:SPB					Memo
SURNAME ▶	Cardone Cardone:lro					
DATE ▶	2/14/72					

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FEB 14 1972

Peter A. Morris, Director, Division of Reactor Licensing

SOUTHERN CALIFORNIA EDISON COMPANY - SAN ONOFRE UNITS NO. 2 & 3
DOCKET NOS. 50-361/362

Adequate responses to the enclosed request for additional information are required before we can complete our review of the subject application. These requests, prepared by the DRS Mechanical Engineering Branch, concern the draft report "Methods of Direct Application of Element Damping" and the material submitted in Amendment 10 of the PSAR within the scope of the review of this branch.

DKT #50-361

Original Signed By
E. G. Case
Edson G. Case, Director
Division of Reactor Standards

Enclosure:
Request for Additional Information
San Onofre 2 and 3

cc w/encl:
E. J. Bloch, DR
S. Hanauer, DR
R. Boyd, DRL
R. DeYoung, DRL
D. Skovholt, DRL
R. Maccary, DRS
D. Lange, DRS
K. Goller, DRL
R. Birkel, DRL
K. Kapur, DRS
J. Knight, DRS

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SURNAME ▶	J. Knight:jm	K. Kapur	D. Lange	R. Maccary	E.G. Case	
DATE ▶	2/11/72	2/11/72	2/11/72	2//72	2/12/72	

ADDITIONAL INFORMATION REQUEST

SAN ONOFRE UNITS 2 & 3

DOCKET NOS. 50-361/362

Seismic System Dynamic Analysis

1. The nonproportional symmetric damping matrix specified in the report "Methods of Direct Application of Element Damping" is based upon approximations that may not produce conservative results. Provide a comparison of the nonproportional damping matrix approach with more accurate substructure modal coupling techniques such as outlined in references 1* and 2* which account for appropriate damping values for soil, moderate damping for the structural model and low damping for piping systems.

2. Provide the basis for the use of a lumped parameter mathematical model with equivalent soil springs in lieu of a finite element model (or equivalent method), including the use of parametric studies which evaluate possible variations in the in situ soil properties (e.g.,moduli, density, stress level, etc.). Include a brief description of the method, mathematical model and damping values (rocking, vertical, translation and torsion) that have been used to consider the soil-structure interaction.

3. Submit a list of the responses obtained from both the modal analysis response spectrum and time history methods, if applicable, for selected points in Category I structures to provide the basis for checking the seismic system analysis.

---

*Reference 1 "Review of Modal Synthesis Techniques and a New Approach" Shou-nien Hou, The Shock & Vibration Bulletin, Dec. 1969 Bulletin 40, part 4.

Reference 2 "Dynamics of Structures", Hunty and Rubinstein, Prentice-Hall Inc. 1st. Edition, 1964.

Seismic Design Input Criteria

1. The proposed seismic design spectra, Figures 2.10-1 and 2.10-2, do not provide an acceptable basis for the seismic design of San Onofre Units 2 and 3. Provide a more acceptable seismic design basis by developing design spectra for the San Onofre site which define the vibratory ground motions of the Safe Shutdown Earthquake and, if applicable, the Operating Basis Earthquake at the elevations of the foundations of the San Onofre Units 2 and 3 structures as required in the Seismic and Geologic Siting Criteria (proposed Appendix A to 10 CFR Part 100). Demonstrate that the final design spectra are developed from an envelope of spectra which are related to the vibratory motions caused by more than one earthquake and reflect the fact that representative response spectra obtained from historic earthquake records show that for 2% damping peak amplification factors are in the range of 2.5 to 5.0 for the period range of 0.15 to 0.5 seconds, and that amplification factors are greater than 1.0 in the period range 0.03 to 0.15 seconds.

2. Provide plots that show a comparison of the smoothed site response spectra and the spectra derived from actual or synthetic earthquake records as applicable for all damping values which will be used in the time history system analyses. Identify the system period intervals at which the response spectra acceleration values were calculated and demonstrate that the period interval used is sufficient to produce accurate spectra that envelope and do not deviate below the smooth response spectra for the site.

Seismic Subsystem Analysis

The use of the floor response spectrum for a single particular elevation may not be sufficiently conservative for the analysis of multidegree-of-freedom systems which are supported at several elevations (See P. 18-156 of Amendment 10 to PSAR). Provide the design criteria and analytical procedures applicable to piping that take into account the relative displacements between piping support points, i.e. floors and components, at different elevations within the structures and between structures.

JAN 31 1972

Peter A. Morris, Director, Division of Reactor Licensing

QUESTIONS RELATING TO PROTECTION AND EMERGENCY POWER SYSTEMS, SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 & 3, DOCKET NOS. 50-361/362

Please include the attached questions among those in preparation for transmittal to the applicant.

Original Signed By
E. G. Case
Edson G. Case, Director
Division of Reactor Standards

ESB-8
DRS:ESB:DFS

cc w/encl:
S. Hanauer, DR
R. DeYoung, DRL
R. Boyd, DRL
D. Skovholt, DRL
K. Goller, DRL
R. Birkel, DRL
V. Moore, DRS
D. Sullivan, DRS

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DRS RF
ESB RF

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SURNAME ▶	SULLIVAN	MOOORE	CASE			
DATE ▶	1/28/72	1/28/72	1/28/72			

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JAN 17 1972

Original Signed By
K. R. Goller

R. C. DeYoung, Assistant Director for PWR's, DRL
THRU: K. R. Goller, Chief, PWR Branch No. 3, DRL

SOUTHERN CALIFORNIA EDISON COMPANY - SAN DIEGO GAS AND ELECTRIC COMPANY
SAN ONOFRE UNITS 2/3 - DOCKET NOS. 50-361 AND 50-362

Enclosed is a summary of the meeting held with the U. S. Geological Survey and Southern California Edison Company and San Diego Gas and Electric Company on January 7, 1972, ~~in Menlo~~ Park, California. The meeting was requested by the applicant. An attendance list is also enclosed.

Original Signed By
R. A. Birkel

Ralph A. Birkel
PWR Branch No. 3
Division of Reactor Licensing

Enclosures:
1. Meeting Summary
2. Attendance List

cc w/encls:
P. A. Morris, DRL
F. Schroeder, DRL
T. R. Wilson, DRL
R. S. Boyd, DRL
D. J. Skovholt, DRL
H. R. Denton, DRL
R. W. Klecker, DRL
DRL/DRS Branch Chiefs
E. G. Case, DRS
R. R. Maccary, DRS
CO (2)
V. H. Wilson, DRL
Meeting Attendees from REG

Memo

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DATE ▶	1/11/72	1/ /72				

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ENCLOSURE NO. 1

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

SAN ONOFRE UNITS 2/3 - DOCKET NOS. 50-361 & 50-362

PRELIMINARY RESULTS OF OFFSHORE GEOPHYSICAL INVESTIGATION

SAN ONOFRE SITE

MEETING OBSERVATIONS

Summary

At the direct request of the applicant a meeting was held with the U. S. Geological Survey - Western Center, Menlo Park, California, with the intent of briefing the USGS on the preliminary results of the offshore San Onofre geophysical investigation. The briefing did not include evaluation of the results of the investigation nor did it present any conclusions.

Discussion

Using charts and maps, the applicant showed the preliminary results of the offshore investigation that was conducted by the Western Geophysical Company in October/November, 1971. The documents presented included offshore contour maps of the ocean bottom, acoustical maps of the basement, and raw data acquisition charts. All of the raw data had not as yet been reduced and thus the documents were all in an incomplete stage. It is the intent of the applicant to have completed results available by mid-January.

Although the applicant appeared to allude at times to reaching conclusions, upon direct questioning they stated that the investigation and results were still open to consideration. It was the impression of both the writer and USGS that the applicant's final conclusion might be that the additional data supports their previous position. We were informed that subsequent to the completion of the current investigation, the seismology for the site will be developed by Stuard Smith and Ron Scott (both members of the SCE Board of Technical Review).

In private discussions with the USGS, the writer stated that it was his observation from the charts and data presented by the applicant that the onshore Newport-Inglewood fault appeared to definitely extend offshore. However, a connection with the Rose Canyon structure was not evident due to incomplete data reduction in this area by the applicant's consultant, Western Geophysical. It was agreed by everyone that the amount and perhaps even the quality of the offshore data which will now be available is outstanding.

The applicant indicated that they expected to conclude the current investigation by mid-January and to provide the staff with a draft of their evaluation by late January.

ENCLOSURE NO. 2

ATTENDANCE LIST

JANUARY 7, 1972, MEETING

SOUTHERN CALIFORNIA EDISON COMPANY

Orlando J. Ortega
Philip J. West
Jack B. Moore
Kenneth P. Baskin
Gail S. Hunt

SAN DIEGO GAS & ELECTRIC COMPANY

Bob Lacy

USGS - MENLO PARK

Holly Wagner
R. Yerkel
J. P. Eaton
Joseph Ziony
Robert E. Wallace

AEC - DRL

R. A. Birkel

DEC 2 1971

Original Signed By
K. R. Goller

R. C. DeYoung, Assistant Director for PWR's, DRL
THRU: K. R. Goller, Chief, PWR Branch No. 3, DRL

SOUTHERN CALIFORNIA EDISON COMPANY - SAN DIEGO GAS AND ELECTRIC COMPANY
SAN ONOFRE UNITS 2/3 - DOCKET NOS. 50-361 AND 50-362

A meeting was held in Bethesda, Maryland with Mr. K. P. Baskin, Southern California Edison Company, on November 22, 1971, for the purpose of delineating pending review items, exclusive of geology/seismology, that require resolution prior to the completion of our evaluation of the SCE and SDG&E request for a construction permit for San Onofre Units 2 and 3. A summary of these items is enclosed.

Staff review of the geology and seismology associated with our evaluation has been suspended pending receipt of additional information (primarily offshore data) which SCE is currently obtaining and evaluating. SCE indicated that a draft of this information should be available for review and discussion with the staff by mid-January, 1972.

SCE indicated that a supplement to its environmental report would be submitted in January, 1972. This submission date appears to be compatible with the overall DREP review schedule for San Onofre. In a related matter, pursuant to California State water quality requirements, SCE will redesign its circulating water system discharge lines (increasing length, separation of lines and revise diffuser) to meet a 4 °F max. temperature rise within 3000 ft. of the discharge point.

Pending satisfactory resolution of the geology/seismology concern and the action review items (enclosure), SCE would desire an ACRS meeting in March/April, 1972.

Original Signed By
R. A. Birkel

Ralph A. Birkel
PWR Branch No. 3
Division of Reactor Licensing

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PWR-3 Reading

Enclosure:
Pending Review Items,
November 19, 1971

cc w/encl: See attached

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SURNAME ▶	RABirkel:esp	KRGoller				
DATE ▶	11/29/71	11/ /71				

cc w/encl:
P. A. Morris, DRL
F. Schroeder, DRL
T. R. Wilson, DRL
D. J. Skovholt, DRL
R. S. Boyd, DRL
H. R. Denton, DRL
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DRL/DRS Branch Chiefs
E. G. Case, DRS
R. R. Maccary, DRS
CO (2)
V. H. Wilson, DRL

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DATE ▶						

ENCLOSURE NO. 1

November 19, 197

SAN ONOFRE 2/3

PENDING REVIEW ITEMS

(Exclusive of Geology/Seismology)

DOCKET NOS. 50-361 & 50-362

1. Tornado design - required to use standard AEC design basis tornado.

2. RCS pressure boundary - use of faulted stress limits requires resolution.

3. System quality group classification - CVCS letdown line requires code B; code D not acceptable for radwaste unless failure is within 0.17 mrem limit.

4. Spent fuel storage and fuel handling system -

   a) tornado design criteria

   b) Safety Guide No. 13 (auxiliary building auto vent)

   c) Requirements - minimum 23 feet of water over spent fuel, impossible to have fuel cask over spent fuel pool, maximum 30 feet fuel cask drop distance, drop of loaded fuel cask anywhere in travel not to effect safety related functions or public health; Class I makeup coolant system.

5. Require automatic protection for core power maldistributions.

6. Compliance with Appendix A, 10 CFR Part 50 (July, 1971), General Design Criteria. (especially GDC 55-57).

7. Conformance with the intent of all applicable portions of Safety Guides 1-18.

8. Adequacy of ECCS (ref: DRL letter, August 2, 1971).

9. Consequences of a single rod withdrawal accident; design should show that this is not anticipated transient, if unable to do so, must include CEA withdrawal prohibit or other.

10. Consequences to reactor of a main steam line break; include steam line restraints and commitment no containment liner damage; requirement for flow restrictors.

11. Requirement for vibration monitoring program (draft Safety Guide).

12. Compliance with Appendices G and H, 10 CFR 50 (fracture toughness and radiation surveillance criteria).

13. Commitment and details required on preliminary design of combustible gas control system.

14. Compliance with flywheel integrity safety guide (No. 14); requirement for underfrequency and undervoltage RC pump trip.

15. Status of dynamic analysis of reactor vessel internals.

16. Reactor cavity pressure during LOCA.

17. Adequacy of cooling water system - ultimate heat sink; general discussion.

18. Compliance with Appendix I, 10 CFR 50 (as low as practicable releases).

19. RCS pressure boundary overpressure protection - require (a) criteria, bases and analysis in PSAR and (b) commitment to include ASME report in FSAR.

20. Adequacy of emergency plan to protect visitors to adjacent California State Park.

21. Plant protection - compliance with Safety Guide No. 17.

22. Effects of boric acid solution on corrosion of RCS pressure boundary (ref: NOK I, Beznau); general discussion.

23. Water chemistry limits including pH control.

24. Consequences of steam generator tube wall thinning, acceptable 1 gpm primary/secondary leak and subsequent steam line break accident.

25. Effects of turbine generator trips with loss of all AC.

26. Effect of three pump operation on accidents analyzed.

27. Failed fuel detection sensitivity vs. number of failed fuel assemblies.

28. CP technical specifications using reference plant and exceptions.

29. Conformance with IEEE codes:

    IEEE-338 (development of test program; periodic testing)
    -334 (environmental testing of electrical motors inside containment, LOCA)
    -344 (testing Class I electric equipment).

30. RHR system requirements (a) isolation valve interlocks and auto closure and (b) method of operation with passive failure at low temperature and pressure.

31. Safety injection tank isolation valve interlocking and auto opening.

32. Compliance with Appendix J, 10 CFR 50 (containment leak rate testing).

33. Adequacy of post-accident monitoring provisions.

32. Verification that personnel are not required to leave control room during LOCA.

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Docket 50-362
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PWR-3 Reading

OCT 14 1971

Original Signed By
K. R. Goller

R. C. DeYoung, Assistant Director for PWR's, DRL
THRU: K. R. Goller, Chief, PWR Branch No. 3, DRL

SOUTHERN CALIFORNIA EDISON COMPANY - SAN DIEGO GAS AND ELECTRIC COMPANY, SAN ONOFRE UNITS 2/3 - DOCKET NOS. 50-361 AND 50-362

Enclosed is a summary of the meeting held with Southern California Edison Company and San Diego Gas and Electric Company on October 5, 1971, in Rosemead, California. An attendance list is also enclosed.

Original Signed By
R. A. Birkel

Ralph A. Birkel
PWR Branch No. 3
Division of Reactor Licensing

Enclosures:
1. Meeting Summary
2. Attendance List

cc w/encls:
P. A. Morris, DRL
F. Schroeder, DRL
T. R. Wilson, DRL
R. S. Boyd, DRL
D. J. Skovholt, DRL
H. R. Denton, DRL
DRL/DRS Branch Chiefs
V. H. Wilson, DRL (2)
E. G. Case, DRS
R. R. Maccary, DRS
R. W. Klecker, DRL
CO (2)
Meeting Attendees from REG
E. Baltz, USGS
J. F. Devine, NOAA

OFFICE ▶	DRL:PWR-3 x7415 RAB	DRL:PWR-3 KRG				Memo
SURNAME ▶	RABirkel:esp	KRGoller				
DATE ▶	10/12/71	10/12/71				

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

ENCLOSURE NO. 1

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

SAN ONOFRE UNITS 2/3 - DOCKET NOS. 50-361 & 50-362

PLANNED OFFSHORE GEOPHYSICAL INVESTIGATION

SAN ONOFRE SITE

MEETING NOTES

Summary

A technical meeting was held with the applicant at the main offices of the Southern California Edison Company in Rosemead, California on October 5, 1971. The purpose of the meeting was to provide the staff and staff consultants with information and understanding of the planned investigations and explorations which Western Geophysical Company will conduct offshore at the San Onofre site to provide additional data and information that can be used to establish appropriate seismic criteria for the San Onofre Station.

Discussion

Dr. E. J. "Buck" Mateker, Vice President, Western Geophysical Company, presented the proposed program, and was the chief spokesman during the meeting. Appendix A presents a summary outline of the proposed program. Seismic reflection technique using the Aquapulse system will be employed to develop common depth point profiling of the offshore areas. Western Geophysical has some 750 miles of reflection profiles of the data aquisition area available and is currently evaluating the data at their Houston, Texas office. In addition, approximately 232 miles of new reflection profiles will be obtained to provide a complete boundary of desired reflection data. The seismic reflection technique and the associated digital data processing system have been used almost exclusively for offshore exploration for the past 4 years with many previous years of good experience applying similar techniques and principles. Properly evaluated results have correlated well with actual conditions encountered by oil companies in drilling.

As a result of discussions with Western Geophysical and the applicant, the staff offered the following comments and suggestions:

1. Running of additional profiles more closely spaced directly offshore from the site north to Newport and south to Oceanwide (and within about 10 miles of the shore) would be desirable. Reduction of the data might not be needed depending on results of presently planned program.

2. Correlate deep (Aquapulse) with shallow (Sparker) data.

3. Cross calibrate old vs. new reflection data (profiles).

4. Consider additional onshore velocity measurement to confirm offshore refraction data.

The offshore data acquisition program is scheduled to start about October 15 and be completed by October 30, 1971. The reduction of the raw data, and interpretation and evaluation of the data is to be completed by early December, 1971. It was mutually agreed that if results of the program were available on the above schedule, that the staff and the applicant could meet in mid-December (December 15/16), to review the results of the investigations and explorations. In addition, the applicant indicated that they hoped to be in a position to discuss their soil-structure program results with the staff in early December (December 7/8).

APPENDIX A

PROPOSED PROGRAM OF OFFSHORE GEOPHYSICAL EXPLORATION

BY WESTERN GEOPHYSICAL COMPANY

FOR SAN ONOFRE UNITS 2 AND 3

1. Obtain Western Proprietary Spec Data Pt I San Diego Area, 1970, and Phase I Outer Banks 750 miles.

2. Acquisition of Data

   a) Seismic Reflection, AQUAPULSE array, Model C guns, Sum 4,1200%, one mile cable, digital recording 232 miles.

   b) Sea magnetometer data, analog recording, 360 miles.

   c) Seismic Refraction, 4 profiles.

   Includes Vessel (Western Crest), crew, and position surveying (Shoran).

3. Seismic Reflection Processing,

   Sum 4,1200% stack, deconvolved before stack; velocity analysis (VELAN) every two miles, 232 miles.

4. Magnetometer Data Processing,

   Regional correction, profile and map presentation, 360 miles.

5. Seismic Refraction Processing, Four Profiles.

6. Velocity Analysis (VELAN), additional located for specific geologic control, approximately 20 VELANS.

7. Seismic Section Full-Waveform Migration, approximately 100 miles.

8. Average Velocity Section with Internal Velocity distribution, approximately 50 miles.

9. Interpretation Seismic and Magnetic Data, Geologic Integration, and Reporting, 1000 miles.

ENCLOSURE NO. 2

ATTENDANCE LIST

SOUTHERN CALIFORNIA EDISON COMPANY

B. R. Laverty
K. P. Baskin
P. J. West
H. B. Ray
G. S. Hunt

SAN DIEGO GAS AND ELECTRIC COMPANY

R. G. Lacy

WESTERN GEOPHYSICAL

E. J. Mateker
H. Murphy

NOAA

J. F. Devine

USGS

E. H. Baltz
H. C. Wagner
J. G. Vedder
J. Ziony
R. Yerkes

AEC - DRL/DRS

R. A. Birkel
R. B. Minogue

AUG 3 1 1971

50-361
50-362

C. I. York, Assistant Director for Engineering
Division of Construction

COST DATA FOR PWR PLANTS

Reference: Memo, dtd August 24, 1971, subject as above, from C. I. York, Division of Construction

With respect to the request contained in your memorandum referenced above, I am enclosing cost information for the plants listed in your memo. In cases where the applicant has updated the financial data since the filing of their license application, I have included that information. However, in some cases, the applicant has not filed any additional financial data since the filing of the initial application. In these cases, I have included the initial license application.

In order to expedite your request, we have not reproduced these documents. I would appreciate it if you would please return them when they have served their purpose.

Original signed by R. C. DeYoung

Richard C. DeYoung, Assistant Director
for Pressurized Water Reactors
Division of Reactor Licensing

Enclosures:
As stated on the attachment

cc: C. Long
A. Schwencer
K. Goller
D. Muller

DISTRIBUTION:
Docket
DRL Reading
PWR-1 Reading
V. Wilson
F. Karas
N. Brown

CRESS	OFFICE ▶	AD:PWR	AD:PWR				
T142	SURNAME ▶	NBrown:mlm	RCDeYoung				
8/31/81	DATE ▶	/ /71	8/3/71				

AUG 3 1 1971

Docket No.	Plant	Unit
50-313 Amendment No. 19, "Application for Operating License"	Arkansas Nuclear No. 1	1
50-334 Amendment No. 7	Beaver Valley	1
50-338 & 50-339 Amendment No. 5 & Amendment No. 11	North Anna	1,2
50-327 License Application	Sequoyah	1,2
50-363 License Application	Forked River	1
50-348 Amendment No. 1	Jos. M. Farley	1,2
50-369 & 50-370 Amendment No. 9	William B. McGuire	1,2
50-361 & 50-362 License Application & Amendment No. 2	San Onofre	2,3
50-376 Letter, dated 3/8/71 from PRWA & Attachment No. 1 to this letter	Aguirre	-
50-382 & 50-383 License Application	Waterford	3
50-390 & 50-391 License Application	Watts Bar	1,2
50-395 License Application	Summer	-

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Docket Files
50-361
-362

AUG 19 1971

NOTE FOR HAROLD L. PRICE
CLIFFORD K. BECK
MARVIN M. MANN
C. L. HENDERSON
STEPHEN H. HANAUER
PETER A. MORRIS

SAN ONOFRE SEISMIC DESIGN BASIS MEETING - FRIDAY, AUGUST 20, 1971

As you know, a meeting has been scheduled on Friday, August 20, in Room P-422, starting at 9:30 a.m. with Southern California Edison representatives to discuss the seismic design basis for San Onofre Units 2 and 3. The following Southern California Edison personnel are scheduled to attend:

W. R. Gould
J. B. Moore
O. J. Ortega
B. R. Laverty
P. J. West

Topics for discussion are given in Enclosures 1 and 2. Hank Coulter and Elmer Baltz, USGS, will present the results of the USGS review to date and cover items 1 through 6 of the "AEC list of topics" (Enclosure 1) and item I of the Southern California list (Enclosure 2). Jim Devine of USC&GS will discuss items 6 through 8 of the AEC list, and item II of the Southern California list. Nate Newmark (who will arrive around noon) will cover item III of the Southern California list, and all AEC consultants will discuss item 9 of the AEC list.

The principal purpose of the meeting is to discuss with Southern California Edison personnel the recommendations of the AEC consultants and the basis for these recommendations.

Original Signed By
E. G. Case

Edson G. Case, Director
Division of Reactor Standards

Enclosures:
1. "Geological and Seismological Considerations Important to Determination of Seismic Design Basis for San Onofre," 8/12/71
2. Southern California Edison's Suggested Areas of Discussion, 8/16/71

OFFICE ▶			DRS:DIR			
SURNAME ▶			Case:jl			
DATE ▶			8/18/71			

SOUTHERN CALIFORNIA EDISON'S SUGGESTED AREAS OF DISCUSSION WHERE FURTHER CLARIFICATION AND/OR STUDY MAY LEAD TO BETTER AGREEMENT OF CAPABILITY OF N-I ZONE OF DEFORMATION OF SITE RESPONSE

I. Geological Considerations

A. Items 1 through 6 of the AEC's list of topics, with special emphasis on differences between San Andreas and the N-I zone of deformation characteristics.

B. Characteristic of any additional investigations which could lead to a better understanding of N-I zone of deformation capability.

II. Seismological and Soils Considerations

A. Items 7 and 8 of the AEC's list of topics, with special emphasis on acceptable methods of arriving at a site acceleration based on any one geologic model.

B. Consideration of specific site data as an acceptable method of arriving at site acceleration.

III. Unique Site Conditions Known to Mitigate Structural Response to Any Earthquake

A. Structural damping, soil damping, and soil structural interaction considerations.

B. Unit 1 forced vibration tests.

C. Structural analysis technique.

D. Credit for soil-structure interaction as it relates to item II. above.

GEOLOGICAL AND SEISMOLOGICAL CONSIDERATIONS IMPORTANT TO DETERMINATION OF SEISMIC DESIGN BASIS FOR SAN ONOFRE

1. Identification of principal onshore faults or other similar geologic structures requiring consideration.

2. Identification of principal offshore faults or other similar geologic structures requiring consideration.

3. Significant characteristics of onshore faults or other similar geologic structures requiring consideration.

4. Results and interpretation of offshore acoustical profiles, including percent of profiles which identified faults and folds along trend.

5. Relationships of major onshore and offshore faults and other similar geologic structures, and bases for relationships.

6. Identification and characterization of faults or other similar geologic structures which are controlling in determination of design basis for San Onofre.

7. Determination of Design Basis Earthquake for controlling fault or other similar geologic structure, and basis for determination.

8. Determination of characteristics of DBE ground acceleration at site, and basis for determination.

   a. Maximum ground acceleration.

   b. Amplification at 2% damping.

   c. Response spectrum shape.

9. Validity and applicability of Pacoima dam record of San Fernando earthquake.

SCE Attendees at San Onofre Meeting Friday, August 20, 1971

William R. Gould
Jack B. Moore
Orlando J. Ortega
Bruce R. Laverty
Phil J. West

Note: HLP attended part of this meeting HLP 8/19

Note

50-361
-362

AUG 19 1971

NOTE FOR HAROLD L. PRICE
CLIFFORD K. BECK
MARVIN M. MANN
C. L. HENDERSON
STEPHEN H. HANAUER
PETER A. MORRIS

SAN ONOFRE SEISMIC DESIGN BASIS MEETING - FRIDAY, AUGUST 20, 1971

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B. R. Laverty
P. J. West

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The principal purpose of the meeting is to discuss with Southern California Edison personnel the recommendations of the AEC consultants and the basis for these recommendations.

Edson G. Case, Director
Division of Reactor Standards

Enclosures:
1. "Geological and Seismological Considerations Important to Determination of Seismic Design Basis for San Onofre," 8/12/71
2. Southern California Edison's Suggested Areas of Discussion, 8/16/71

Rec'd Off. Dir. of Reg.
Date 8/19/71
Time 9:00

8/12/71

GEOLOGICAL AND SEISMOLOGICAL CONSIDERATIONS IMPORTANT TO DETERMINATION OF SEISMIC DESIGN BASIS FOR SAN ONOFRE

1. Identification of principal onshore faults or other similar geologic structures requiring consideration.

2. Identification of principal offshore faults or other similar geologic structures requiring consideration.

3. Significant characteristics of onshore faults or other similar geologic structures requiring consideration.

4. Results and interpretation of offshore acoustical profiles, including percent of profiles which identified faults and folds along trend.

5. Relationships of major onshore and offshore faults and other similar geologic structures, and bases for relationships.

6. Identification and characterization of faults or other similar geologic structures which are controlling in determination of design basis for San Onofre.

7. Determination of Design Basis Earthquake for controlling fault or other similar geologic structure, and basis for determination.

8. Determination of characteristics of DBE ground acceleration at site, and basis for determination.

   a. Maximum ground acceleration.

   b. Amplification at 2% damping.

   c. Response spectrum shape.

9. Validity and applicability of Pacoima dam record of San Fernando earthquake.

8/16/71

SOUTHERN CALIFORNIA EDISON'S SUGGESTED AREAS OF DISCUSSION WHERE FURTHER CLARIFICATION AND/OR STUDY MAY LEAD TO BETTER AGREEMENT OF CAPABILITY OF N-I ZONE OF DEFORMATION OF SITE RESPONSE

I. Geological Considerations

A. Items 1 through 6 of the AEC's list of topics, with special emphasis on differences between San Andreas and the N-I zone of deformation characteristics.

B. Characteristic of any additional investigations which could lead to a better understanding of N-I zone of deformation capability.

II. Seismological and Soils Considerations

A. Items 7 and 8 of the AEC's list of topics, with special emphasis on acceptable methods of arriving at a site acceleration based on any one geologic model.

B. Consideration of specific site data as an acceptable method of arriving at site acceleration.

III. Unique Site Conditions Known to Mitigate Structural Response to Any Earthquake

A. Structural damping, soil damping, and soil structural interaction considerations.

B. Unit 1 forced vibration tests.

C. Structural analysis technique.

D. Credit for soil-structure interaction as it relates to item II. above.

JUN 8 1971

FILES
THRU: K. R. Goller, Chief, Branch No. 3, DRL

SAN ONOFRE UNITS 2 & 3, ACRS SUBCOMMITTEE MEETING TO DISCUSS GEOLOGY AND SEISMOLOGY - DOCKET NOS. 50-361 & 50-362

Jim Hard, ACRS Staff Assistant, phoned on June 7, 1971 to inform the staff that Dr. Seiss would like to schedule an ACRS Subcommittee meeting on a mutually agreeable date between June 14 and July 7. The purpose of the meeting is to discuss San Onofre geology and seismology. It was requested that all staff consultants (USGS, NOAA and Newmark & Hall) be available at the meeting. ACRS consultants will also be present. The meeting date will be determined by the availability of the attendees.

Original Signed by
R. A. Birkel

Ralph A. Birkel
PWR Branch No. 3
Division of Reactor Licensing

cc: R. C. DeYoung
R. Minogue

OFFICE ▶	DRL:PWR-3	DRL:PWR-3				Memo
SURNAME ▶	x-7243 RAB RABirkel:tls	KRG KRGoller				
DATE ▶	6/7/71	6/8/71				

June 4, 1971

FILE

TELECON WITH K. P. BASKIN, SOUTHERN CALIFORNIA EDISON COMPANY
SAN ONOFRE UNITS 2 & 3 - DOCKET NOS. 50-361 & 50-362

Original Signed by
K. R. Goller

On June 3, 1971, I contacted K. P. Baskin to inform his of the postponement of the San Onofre 2/3 ACRS Subcommittee meeting originally scheduled by the Subcommittee chairman, Dr. Seiss, for Tuesday, June 8, 1971. This postponement also removed the San Onofre 2/3 application review from the full ACRS Committee June schedule (originally scheduled for June 10, 1971). The postponement resulted from the fact that the staff had not as yet concluded its evaluation of the site geology and seismology and thus had not established a ground acceleration value for the DBE. The staff consultants have indicated that based upon the site structural geology a 'g' value of approximately 0.75g may be appropriate for the DBE. The applicant stated that he would evaluate the implications of the latter on the design of the plant and its effect on the San Onofre application and would subsequently contact the staff.

Original Signed by
R. A. Birkel

Ralph A. Birkel
PWR Branch No. 3
Division of Reactor Licensing

OFFICE ▶	~~DRL:PWR-3~~ RAB					
SURNAME ▶	RABirkel:tls					
DATE ▶	6/4/71					

MAY 2 6 1971

P. A. Morris, Director, DRL

SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3, DOCKET NOS. 50-361 AND 50-362

The PSAR submitted by the subject applicant has been reviewed and evaluated by the DRS Mechanical Engineering Branch. A final evaluation of the material within the scope of review of this Branch is enclosed. Tentative conclusions, for which confirmation is still required are enclosed in parentheses; the material in brackets provides a summary of actions to be taken to resolve issues still open at this final evaluation review stage.

Original signed by
E. G. Case

Edson G. Case, Director
Division of Reactor Standards

Dkt 50-361

Enclosure:
Final Evaluation - Mechanical for
San Onofre 2 and 3

cc w/encl:
S. Hanauer, DR
R. Boyd, DRL
R. DeYoung, DRL
D. Skovholt, DRL
R. Maccary, DRS
K. Goller, DRL
D. Lange, DRS
R. Birkel, DRL
R. Kirkwood, DRS
K. Wichman, DRS
J. Knight, DRS

CRESS OFFICE ▶	DRS:MEB	DRS:MEB	DRS:MEB	DRS:A/D	DRS:DIR	
T-45 SURNAME ▶	KWichman:en	RKirkwood	DFLange	RMaccary	EGCase	
R1-16:en DATE ▶	5/12/71	5/ /71	5/25/71	5/2?/71	5/25/71	

SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 & 3

DOCKETS NOS. 50-361 AND 50-362

System Quality Group Classifications

The applicant has applied a system of code classification groups to those pressure-containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety. These classification groups, Nuclear Code Classes A, B and C, and Non-Nuclear Code Class D generally correspond to the tentative code classification groups A, B, C and D developed by the staff. The codes applicable to the components in each of the classification groups developed by the staff are identified in Table CS-1, "Summary of Codes and Standards for Components of Water-Cooled Nuclear Power Units".

The applicant and the staff are in general agreement on the application of the code classification groups for the reactor coolant pressure boundary and the majority of those fluid systems important to safety.

[The applicant differs in classification with the staff for those portions of the Chemical and Volume Control System (CVCS) that comprise the reactor coolant letdown loop. The applicant's Code Class C of the CVCS letdown loop would be acceptable provided the concept of system shrink during cooldown as a means of reducing system volume concurrent with boron injection is acceptable to DRL, instead of the feed and bleed mode of operation normally used during shutdown.]

(The applicant's Code Class D for the Coolant Radwaste System, Boric Acid Recycle System and the Filter/Ion Exchanger loop of the Fuel Pool Cooling System is acceptable only if the applicant provides documentation satisfactory to DRL that the failure of components in these systems would not result in calculated potential exposures in excess of 0.17 rem whole body (or its equivalent to parts of the body) at the site boundary or beyond.)

TABLE CS-1 Summary of Codes and Standards for Components of Water-Cooled Nuclear Power Units 2/12/71

Component	Code Classifications Group A	Group B	Group C	Group D
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, Class A	ASME Boiler and Pressure Vessel Code, Section III, Class C	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1 or Equivalent
0-15 Psig Storage Tanks	–	API-620 with the NDT Examination Requirements in Table NST-1, Class 2	API-620 with the NDT Examination Requirements in Table NST-1, Class 3	API-620 or Equivalent
Atmospheric Storage Tanks	–	Applicable Storage Tank Codes such as API-650, AWWAD100 or ANSI B 96.1 with the NDT Examination Requirements in Table NST-1, Class 2	Applicable Storage Tank Codes such as API-650 AWWAD100 or ANSI B 96.1 with the NDT Examination Requirements in Table NST-1, Class 3	API-650, AWWAD100 or ANSI B 96.1 or Equivalent
Piping	ANSI B 31.7, Class 1	ANSI B 31.7, Class II	ANSI B 31.7, Class III	ANSI B 31.1.0 or Equivalent
Pumps and Valves	Draft ASME Code for Pumps and Valves Class I. See Footnote (a)	Draft ASME Code for Pumps and Valves Class II. See Footnote (a)	Draft ASME Code for Pumps and Valves Class III	Valves – ANSI B 31.1.0 or Equivalent Pumps – Draft ASME Code for Pumps Valves Class III or Equivalent

FOOTNOTE:

(a) All pressure-retaining cast parts shall be radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant examination may be substituted. Examination procedures and acceptance standards shall be at least equivalent to those specified in the applicable class in the code.

Reactor Coolant Pressure Boundary - Design

The reactor coolant pressure boundary (RCPB) will be designed as a Class I (seismic) system to withstand the normal loads of mechanical, hydraulic, and thermal origin, including anticipated transients, and the operational basis earthquake within the stress limits of the applicable component codes.

The stress criteria proposed for the design of RCPB components for the combined loads resulting from the Design Basis Earthquake and the Design Basis Accident is presented in Table B.3-1 of the PSAR. (We find that the criteria for the faulted operating condition category exceed appropriate code allowable stress limits. Although the applicant has documented his intent to comply with the limits of Case 70 of Interpretations of Code for Pressure Piping for piping designed to Class I requirements of ANSI B31.7, the "faulted condition" stress limits for vessels, pumps and valves are still in question. We expect, however, to achieve satisfactory resolution of this issue prior to the ACRS meeting.)

(Paragraph I-701.5.4 of the ANSI B31.7 Nuclear Power Piping Code requires that piping shall be supported to minimize vibration and that the designer is responsible by observation under startup or initial operating conditions to assure that vibration is within acceptable levels. Because of this requirement, and the recent failures that occurred at operating plants which were attributable to excessive vibration levels, we have asked the applicant to conduct a vibration operational test program to verify that the piping and piping restraints within the RCPB have been designed to withstand dynamic effects due to valve closures, pump trips, etc. The applicant has not responded to our request as of this date, however, we anticipate no difficulty in obtaining confirmation of his intent to comply with this code requirement.)

Reactor Internals - Design

For normal design loads of mechanical, hydraulic, and thermal origin, including anticipated plant transients and the Operational Basis Earthquake, the reactor internals will be designed to function within the stress limit criteria of Article 4, Section III of the ASME Boiler and Pressure Vessel Code.

All internals components are designated as Class I seismic items and will be designed to withstand loads resulting from a combined Design Basis Earthquake and Loss-of-Coolant Accident. Membrane strain limits for the internals under this combined load will correspond to an elastically calculated stress of approximately two-thirds of the specified minimum tensile strength for the applicable material at temperature. The stress results calculated for the combined Design Basis Earthquake and Loss-of-Coolant Accident indicate that the CEA shrouds in the first row nearest the reactor vessel outlet nozzles slightly exceed the estimated stress at assumed failure. However, the remaining CEA shrouds and all other internal components important to safety are within 55 percent or less of the stress at failure assuring that core cooling and reactor shutdown capability will not be impaired.

To prevent liftoff of the core support plant at maximum cold flow conditions during pressure surges, a holddown device will be incorporated into the upper end fitting of the fuel assembly. This device consistent of a spring loaded movable spider which acts on the underside of the fuel alignment plate. The spider and end fitting will be fabricated of 304 stainless steel and the coil springs will be constructed of Inconel X750 material. The springs will be subject to their maximum stress range for a plant heatup or cooldown which is estimated to occur less than 50 times over the life of a fuel assembly. The design criteria limit the maximum stress to below the yield strength of the spring material. Since fuel assembly hold-devices have not been utilized in previous designs by the nuclear steam system supplier, Combustion Engineering, the most severe cyclic conditions will be tested as a part of the fuel assembly development program.

Class I (Seismic Design) Safety Related Systems

We and our seismic consultants have evaluated the analytical procedures which will be used to calculate the seismic loads on Class I (seismic design) systems, components, and equipment and find these procedures to be acceptable. The quality assurance requirements imposed by the applicant for these items comply with the provisions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants," of 10 CFR 50.

All Class I systems, components, and equipment will be designed to sustain the Operational Basis Earthquake within the appropriate code allowable stress limits. (However, the applicant has been requested to reconsider his response to our question 4.14 in Amendment 4 to the PSAR, which indicates an intent to apply faulted operating condition category stress limits to systems and components outside of the reactor coolant pressure boundary for the combination of loadings which include the Design Basis Earthquake. We have reason to believe that the applicant will submit an acceptable response in view of our recent discussions of this issue.)

Seismic Input

[The proposed seismic design spectra for the Operating Basis Earthquake (OBE) produce a peak amplification factor of approximately 2.2 for 2 percent damping. However, analyses of historic seismic records have indicated amplification factors in the range of 2.5 to 4.5 in the period range of 0.15 to 0.5 seconds. The seismic design spectra for the San Onofre site should include more appropriate amplification factors based upon considering the effects of distance between the seismic disturbances and the site on the predominant periods and historic seismic records. In addition, the criteria used to determine the time histories associated with the seismic design spectra and the enveloping technique employed have not been submitted for our review. As a consequence, we and our seismic consultants conclude that the seismic input criteria proposed by the applicant do not provide an acceptable basis for seismic design of San Onofre Units 2 and 3. In view of the protracted discussions we have had with the applicant, we expect some difficulty in resolving this item prior to the ACRS meeting for this plant.]

Seismic System Dynamic Analyses

Modal response spectrum multi-degree-of-freedom and normal mode-time history methods are used for all Category I structures, systems, and components. Governing response parameters will be combined by the square root of the sum of the squares to obtain the modal maximums when the modal response spectrum method is used. Floor spectra inputs to be used for design and test verification of structure, systems and components are generated from the normal mode-time history method. (A vertical seismic-system dynamic analysis is being employed in lieu of a constant vertical load factor for all structures, systems and components with natural frequencies greater than 20 Hz; however, in accordance with established practice, we have requested that the applicant use the more appropriate value of 33 Hz. We have also asked the applicant to provide a more detailed description of the seismic analysis methods which will be used for piping systems and to justify his proposed procedure for combining the responses for closely spaced modal frequencies. We have reason to believe that we can resolve these remaining issues which are still outstanding before the ACRS meeting for San Onofre 2 and 3.)

Seismic Instrumentation

(The seismic instrumentation which was proposed in the PSAR is not in accordance with the intent of Safety Guide Number 12, "Seismic Instrumentation." We have asked the applicant to comply with the intent of Safety Guide Number 12 and we expect to receive an acceptable response to our request before the ACRS meeting.)

Reactor Internals - Dynamic System Seismic, Operating, and LOCA Analysis

Operating dynamic loads for design will be determined on the basis of analyses performed for similar design plants and confirmed by vibration measurements made on prototype plants. Eigenvalve and eigenfunction calculations based upon the ASYM-2 shell-of-revolution computer program and the STARDYNE finite element computer program will provide information as to the frequencies and associated mode shapes that may be excited in the internals. Seismic loading in the reactor internals will be determined by means of a modal response spectrum multi-degree-of-freedom time history system analysis.

For the Loss-of-Coolant Accident (LOCA) the FLASH computer program will be used to define the flow transient and the WHAM program will determine the corresponding dynamic pressure load distribution. (It is not clear whether the "ring" modes of vibration, which have been shown to provide significant response loadings on similar reactor internals configurations, will be as adequately considered in the dynamic analysis of the internals under LOCA conditions. We have asked the applicant to respond to this concern. We expect that this concern will be resolved prior to the ACRS meeting for this plant.)

Reactor Internals - Vibration Control

As part of previous design efforts by Combustion Engineering, an extensive vibration test program is in progress at the Palisades Plant. The applicant states that no vibration test program is planned for this plant pending the results of the Palisades tests. A comparison of plant characteristics between Palisades and San Onofre reveals significant changes in internals design resulting from differences in coolant flow velocities and length of the reactor vessel. The effect of these differences possibly invalidates extrapolating the Palisades results to this application. Since San Onofre Unit 2 constitutes a prototype reactor; i.e., the initial reactor that is representative of a group of reactors of the same design, size, and configuration, we believe a commitment to perform a program to confirm the satisfactory vibration performance of the San Onofre reactor internals should be made by the applicant prior to issuance of the Construction Permit. We have therefore requested that the applicant perform a preoperational vibration test program within the intent of guidelines considered appropriate for a prototype reactor.

1. A vibration test program for the prototype reactor should include:
   a. a brief description of the vibration test program, including instrumentation type and location,
   b. the expected numerical values of the response of the reactor internals and the anticipated forcing functions, under all flow modes of normal reactor operation,

c. the acceptance standards and the permissible deviations from these standards, and

d. the bases upon which the response, the forcing functions and the permissible deviations were established.

2. A vibration test program should be implemented during the preoperational functional testing program to measure the response[1/] of the reactor internals in order to determine the flow-induced forces and the related dynamic forcing functions for all significant modes of normal reactor operation. The data obtained by these measurements on reactor internals should be sufficient to verify that the steady state and cyclic stresses in the components, as determined by analyses, are within the acceptable design limits set forth in the design specifications and code requirements and that the results meet the acceptance criteria of the vibration test program.

3. The extent of measurements included in a vibration test program should be determined on the basis of the design and configuration of those structural elements of the reactor internals important to safety and the adequacy of theoretical and empirical data used in their design.

1/ Frequency and magnitude of vibration (in terms of displacements, velocities and accelerations).

The type of vibration test instrumentation used, the number of measurement devices within the reactor should be sufficient to determine all significant vibrational modes and characteristics of the reactor internals.

4. After the reactor internals have been subjected to the significant flow modes of normal reactor operation, the reactor internals should be removed and visual or surface examinations of reactor internals should be conducted to detect any evidence of excessive vibrations, and the presence of flaws or wear induced by unanticipated vibration. These examinations should be conducted at all major loadbearing structural elements whose failure could adversely affect integrity of the reactor internals, and at all areas of lateral, vertical and torsional restraints provided within the reactor vessel.

5. A summary of the results obtained from the vibration tests should include:

   a. a description of any differences from the specified vibration test program, instrumentation, reading anomalies and instrument failures,

   b. a comparison between measured values[2/] and the values predicted for the design of the reactor internals,

---

2/ In areas where measurements to determine forcing functions cannot be obtained practically by means of pressure transducers, the forcing functions may be calculated from the measured responses of other areas and the derived vibrational characteristics of the reactor internals. The values of the forcing functions computed from the response and reactor internal vibrational characteristics should be compared with the values of forcing functions used in the design.

c. an evaluation of measurements or observations that exceed acceptable limits or that were unanticipated, and the disposition of such deviations, and

d. a certification by the responsible engineer having authority over the conduct of the vibration test program that the test results documented are correct and in accordance with the actual measurements.

[As a result of our request outlined above, a lengthy discussion with the applicant ensued. The applicant has shown reluctance to provide a commitment to conduct a confirmatory vibration test maintaining that the Palisades vibration test data can be applied through scaling to the San Onofre Units. In light of the applicant's position on this issue, we expect difficulty in resolution of this item before the ACRS meeting for the San Onofre Station.]

Docket
50-362
50-361

MAY 7 1971

Files

SAN ONOFRE 2/3 - SEISMIC SCRAM DISADVANTAGES

On April 22, 1971, Jim Hard, ACRS Staff called to transmit a request from Dr. Seiss, who wished to know the disadvantages of installing a seismic scram (in a nuclear power station).

The following information was obtained from the applicant in a telecon with Ken Baskin on 5/3/71.

Disadvantages of a Seismic Scram

1. Overall gird stability and ability to provide needed electrical power during emergency periods would be jeopardized if many nuclear power stations were shutdown due to a seismic scram.

2. Applicant feels that accelerograph triggers are not reliable and would result in many spurious scrams.

3. A seismic scram on peak acceleration and not on time history information results in tripping the unit when equipment is not near design stresses.

4. If plant is operating at or near design point, a seismic trip would place additional unnecessary stresses on the plant equipment.

5. The secondary plant is designed for a ground acceleration of 0.2 g which is considerably less than the primary plant. Thus, the secondary plant although without automatic vibration protection, would cause the shutdown of the primary system by manual operator action whenever excessive vibration (on the turbine or other component) is observed.

Original Signed by
R. A. Birkel

Ralph A. Birkel
Division of Reactor Licensing

cc: RCDeYoung, DRL
KRGoller, DRL
SDMacKay, DRL

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Moms

MAY 3 1971

Peter A. Morris, Director
Division of Reactor Licensing

SAN ONOFRE 2 & 3 - DOCKET NOS. 50-361/362 - NUCLEAR DESIGN

The enclosed is submitted for inclusion in your report to the ACRS on San Onofre Units 2 and 3.

Original Signed by
E. G. Case

Edson G. Case, Director
Division of Reactor Standards

Enclosure:
Nuclear Design Review

cc w/encl:
S. Hanauer, DR
R. DeYoung, DRL
K. Goller, DRL
R. Birkel, DRL

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DATE ▶	4/28/71	4/29/71	4/30/71			

MATERIAL FOR SAN ONOFRE 2 & 3 ACRS REPORT

NUCLEAR DESIGN

The basic nuclear design for San Onofre 2 and 3 is the same as that used in earlier powerplants designed by Combustion Engineering. In many respects, therefore, acceptability of the design follows from the evaluations of the earlier powerplants. The San Onofre 2 and 3 reactors, however, are specifically designed for higher power operation, and our evaluation of the nuclear design has primarily been addressed to this factor.

The increase in power level to 3390 Mwt for San Onofre 2 and 3 from 2440 Mwt for earlier powerplants, such as Hutchinson Island, is the result of several design changes. Two of these involve the nuclear design. One is an increase in the active core length from 137 inches to 150 inches. This 9.5 percent increase results in the same amount of increase in heat transfer surface area and also in the power level for constant average core power density. Nuclearly, this change would be expected to affect axial xenon stability and, possibly, axial peaking factors.

The other change involving the nuclear design is reduction of the design limit for the rod radial nuclear peaking from 1.76 for Hutchinson Island to 1.55 for San Onofre 2 and 3. Material submitted by the applicant indicates the basis for this reduction is twofold: removal of the provision for operation with a xenon override control rod bank in the core, and fuel management studies which show that lower peaking factors can be obtained than those used in the design of earlier reactors.

The xenon override control bank formerly provided consisted of a group of control assemblies allowed to remain in the core during normal full power operation. These assemblies could be removed to provide reactivity to override xenon buildup following a shutdown and restart. The particular assemblies assigned to this bank were variously redefined at intervals during the core life to distribute burnup. Operation with control assemblies in the core at or near full power produces distortions in the power distribution, and an allowance for this must be made in the peaking factor limits. In the San Onofre 2 and 3 design these xenon reactivity transients will be compensated for by adjustments of the soluble boron concentration, not by control assemblies.

Number of Control Assemblies

Removal of the requirement for the xenon override control group and calculation of control assembly requirements and worth have led the applicant to reduce the number of control assemblies provided from 85 in the initial PSAR submittal to 69.Eight of these are part length assemblies, provided for the control of the axial power distribution. The full length assemblies are predicted to have a worth in excess of 8% $\Delta k/k$. The control assemblies each contain five rods as usually employed in Combustion Engineering designed reactors. There are no longer any double control assemblies attached to a single drive. We find it reasonable for the present stage of the reactor design that the number of assemblies can be reduced for the reasons indicated. We will have the benefit of control worth data in reactors preceding San Onofre 2 and 3 and our operating license review to alter this position, if necessary.

Nuclear Peaking Factors

Our major concern in the review of the nuclear design for the San Onofre 2 and 3 application has been the conservatism of the nuclear design limit peaking factors. The rod radial factor of 1.55 is obviously of greatest concern because it has been reduced from 1.76 in earlier CE designs, but the axial factor of 1.68, although not reduced, is also of concern because it is the combination of these factors which determines the local thermal design limits. The design limit peaking factors are assumed initial conditions for various anticipated transient and accident analyses, including loss-of-coolant accidents, and may not be exceeded at steady state full power for these analyses to be valid. We have not required provision for continuous monitoring of normal peaking factors in PWR designs, but only for asymmetries. The fundamental shape is evaluated by calculations for all expected conditions which influence the degree of peaking, and the design limit peaking factors are determined by the envelope, plus margin, of these predictions. There are frequent (at least, monthly) in-core measurements verifying the power distributions.

With regard to asymmetrical power shapes, to which the ex-core detectors are sensitive, our evaluation involves the ability to sense and control the distribution so that the design peaking factors are not exceeded, or, in the case of potential large power maldistributions, cannot reach the level of damaging fuel.

To supplement the material supplied by the applicant, we asked a number of questions concerning the ability to predict, measure, monitor, and control the power distribution sufficiently well to ensure that the objectives indicated above are attainable in the San Onofre 2 and 3 reactors, or if not, whether additional design provisions might alleviate our concerns. Because the design is not final, some of the information we requested was not produced in response to our questions. We have met with the applicant and at this time have his commitment to submit further material which we agree will assure that the reactors can be designed to operate safely at the rated power level.

The applicant is analyzing operation with the maximum allowable control assembly insertion of 0.2% $\Delta k/k$ in the core to determine the potentiality for power peaking toward the top of the core as a result of burnout in the lower half of the core and likely absence of part length rods in the core early in life. He expects, and we agree, that this small insertion will not lead to a significant tendency for a large power peak toward the top of the core. It will be necessary to determine that this situation has not altered unfavorably at the time of operating license application.

The applicant has provided the following information concerning thermal limits as a function of shape and location of axial power peaks:

DNBR FOR THREE AXIAL SHAPES

Axial Shape	Peak in Lower Half	Symmetrical	Peak in Upper Half
Axial Peaking Factor	1.68	1.47	1.68
Rod Radial Peak	1.55	1.56	1.44
DNBR at 123.3% Power	1.30	1.30	1.30

This shows that the design peaking factors of 1.68 axial and 1.55 radial, or more properly their product, the local nuclear peaking factor, is not limiting if the axial power shape is symmetrical or peaked toward the top of the core. Even lower maximum local peaking factors have the same thermal limit for very flat axial power shapes. Final evaluation that predicted shapes and proposed modes of operation do not lead to power distributions exceeding thermal design limits will have to be made at the operating license stage.

The applicant has provided, and will furnish additional, information concerning the present ability to predict rod radial nuclear peaking factors. These indicate, for a core the size of San Onofre 2 and 3 that the design limit rod radial peaking factor of 1.55 is attainable. Maximum calculated values of this quantity do not exceed 1.4, for the analyses presented. It is also stated that predicted values should not exceed the value of 1.4 to allow for calculational errors and the minor operational variations from the effects of xenon redistribution and control rods. Larger rod radial nuclear peaking factors (1.55) occur in the radial planes encompassed by part length rods, but the axial factor drops sufficiently that the rod radial factor with part length rods present is not a design constraint. The radial power distribution analyses do not include thermal-hydraulic feedback effects on the power distribution, which is conservative. We find the techniques used by CE for these analyses satisfactory. Further information has been promised on this application showing that peaking around control rod thimble "water holes" is treated adequately. We find the information described adequate to provide reasonable assurance that the reactors can be designed with the claimed rod radial nuclear peaking factor limit, but again with the reservation that further evaluation is required for the operating license review.

Part of the reason further evaluations at the operating license stage are suggested above is that much more data from operating reactors is required to assure confidence that power distributions and peaking factors can be predicted accurately. Neither the applicant nor CE is able to commit any particular reactor expected to be operating before the San Onofre 2 and 3 FSAR is received to furnish such data, but both agree that more confirmation of predictive techniques must be furnished for the operating license.

Power Maldistribution Detection and Control

Known sources for possible power maldistributions in San Onofre 2 and 3 are misaligned control rods, fuel loading errors, axial xenon instability and azimuthal xenon instability. The nuclear instrumentation system presently includes the usual split ex-core detectors and 45 strings of six local detectors/string of self-powered in-core detectors.

Automatic protection is provided to maintain a DNB ratio above 1.3 for a single unplanned control assembly in the core, as in earlier CE designed powerplants. Lesser misalignments of full or part length control assemblies are detectable with position indicators and, except for a central rod, tilt indications and alarms from the ex-core detectors. We find this acceptable, although further evaluation that lesser misalignments do not require automatic protection is required the operating license stage.

The applicant considers fuel loading or enrichment errors incredible. Loading or gross enrichment errors should be detectable with the in-core detectors. We plan to require in-core maps at each refueling for such detection. The problems of undetectable pellet enrichment errors are, of course, common to the entire nuclear industry and will be solved on generic basis.

CE currently predicts that the core could exhibit instabilities with respect to axial xenon oscillations at some point in the burnup cycle and is marginally stable against azimuthal xenon oscillations. Provision has therefore been made to control axial oscillations, and it is claimed that azimuthal oscillations also can be controlled. While we expect, with CE, that the final reactor can be demonstrated to be stable azimuthally, the fact that analyses cannot be performed on the final design now, and the absence of detailed data from operating reactors on abilities and performance of the nuclear instrumentation system have prevented the applicant from providing analyses convincing us that no safety problems can arise from control of either form of xenon oscillations. He has therefore agreed to submit responses to our questions on this subject containing commitments that automatic protection would be provided against fuel damage from axial and azimuthal power maldistributions unless he can show conclusively that control of either cannot lead to fuel damage.

We consider alarms normally provided on axial and quadrant power tilts to be adequate protection against power maldistribution problems from unknown sources.

April 29, 1971

K. Goller, Chief, PWR No. 3, Division of Reactor Licensing

INPUT FOR ACRS REPORT ON SAN ONOFRE UNITS NOS. 2 and 3
DOCKET NOS. 50-361, 362

Attached is our input to the ACRS Report. It does not include the geology and seismology, foundation engineering and tornado design sections since these are still under review by management.

Richard P. Grill, Chief
Site Safety Branch
Division of Reactor Licensing

Enclosure
Input for ACRS Report
on San Onofre - Units #2-3

cc w/enclosure:
R. DeYoung
R. Burkel

OFFICE ▶	DRL	DRL				
SURNAME ▶	Chuck Ferrell C.Ferrell:mj	RPG R.P.Grill				
DATE ▶	4/29/71	4/ /71				

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

2.0 SITE AND ENVIRONMENT

2.1 Site Description

The site for San Onofre Units Nos. 2 and 3 is an 84 acre tract of land in San Diego County, California, on the coast of the Pacific Ocean. These reactors are to be located adjacent to San Onofre Unit No. 1 which is also a nuclear generating station. The site is approximately 62 miles southeast of Los Angeles and 51 miles northwest of San Diego. It is located entirely within the Camp Pendleton Marine Corps Base on the northwest end of its 18 mile shoreline. The property upon which these plants will be built is being leased from the United States Government until May 12, 2023. Approximately 98 percent of Camp Pendleton is unimproved and devoted to intermittent practice maneuvers, recreation areas and storage of equipment and supplies. There are no missile sites within a radius of at least 10 miles from the plant. The nearest privately owned land is approximately 2.5 miles from the site. The nearest sizable community center is San Clemente (population 18,000) located five miles from the site. The nearest population center of 25,000 people or more is Oceanside, a distance of 17 miles to the southeast.

The population of Camp Pendleton which surrounds the San Onofre site is extremely variable and is not expected to exceed 40,000. The applicant has indicated that no military personnel will be quartered closer than two miles from the plant site. The principal administrative and main personnel housing areas are located

12 to 15 miles to the southeast. The applicant has defined the minimum exclusion distance from either of the two proposed reactor centerlines as 800 meters and the low population zone distance as 3.0 miles. The population of the low population zone for the year 1970 is 360. The estimated population in the year 1980 is 500 people. The California Parks and Recreation Department is planning a state park which will abut the south property line of the site. The applicant has been requested to indicate how this park, if constructed, will be factored into the emergency plan for this facility.

2.2 Meteorology

The applicant has provided five years of onsite data from meteorological instrumentation located on a 64 foot high pole about 500 feet northwest of the reactor site. The applicant has evaluated these data using a horizontal sigma-theta type method and determined that the worst conditions that could occur five per cent or more of the time correspond closely to a Pasquill Type E condition with a wind speed of 2.2 meters per second. Our evaluation of the five years of data indicates a cumulative 5% occurrence $\frac{X}{Q}$ value of 1.5×10^{-4} sec M^3 which corresponds to a Pasquill Type E condition with a wind speed $\bar{\mu} = 2$ m/sec. The Feb. 1, 1966 report of ESSA on Unit No. 1 confirmed these conditions, however, since that time ESSA and the staff have recognized Δt measurements are needed to determine Pasquill conditions during periods of low wind speed. The applicant has been requested to provide Δt

measurements over at least 100 foot differences in elevation in order that we may be able to determine the Pasquill conditions during periods of low wind speed.

2.3 Geology and Seismology

(LATER)

2.4 Foundation Engineering

(LATER)

2.5 Hydrology

Units Nos. 2 and 3 at the San Onofre facility will draw cooling water from the Pacific Ocean and return it through 2600 foot long pipe lines at about 80,000 gallons per minute. The station site located at 30 feet MLW will provide sufficient height against against high tide and wave action as well as from tsunamis (seismic sea waves). Our consultant, the U. S. Coast and Geologic Survey at the time of the construction permit for Unit No. 1, indicated that a 28 foot high sea wall will provide adequate protection against flooding from this seismic event.

The plant drainage system is based on a one in 100 year storm. The plant however has sufficient freeboard to protect critical components against flooding due to a storm of maximum probable precipitation intensity.

2.6 Environmental Monitoring

Preoperational monitoring for San Onofre Unit No. 1 was initiated in November, 1964 to determine background radioactivity levels in the area of the site. The operational monitoring for Unit 1 began in June, 1967 and continues to the present. An evaluation of the data collected from these programs show only normal variations in the background radioactivity levels. Film packets have been installed at appropriate locations around the site to monitor gross airborne activity. These films are changed on a quarterly basis. Radioactivity particulates in the air are monitored by two air sampling stations, one located four miles to the north in San Clemente and the other two miles east of the plant. The thyroid glands of grazing animals such as jack rabbits, will be monitored for radioactive iodine to determine the degree of possible ground contamination due to a release of activity from the plant. Tap water, open reservoir water from San Clemente and inseason crops will be monitored for radioactivity. In the marine environment, such samples as kelp, shellfish, clams, lobsters, abalone, bass and halibut will be sampled regularly. Tritium analysis of both land and marine samples will be conducted. The applicant has consulted with various State and Federal agencies in establishing this program. Although the applicant's environmental monitoring program for Unit 1 is consistent with those of other operation nuclear power stations, the monitoring program for the operations of all 3 units will need to be modified considerably to conform with the recommendations of AEC's Safety Guide (which is now in draft stage) in the technical specifications at the time the operating licenses are issued for these plants.

San Onofre Unit No. 1 is now being reviewed for a full-term operating license. The technical specifications for this plant will be modified to conform to the recommendations of the monitoring safety guide. This data will be available considerably in advance of the time of the operating license review of Units 2 and 3. We, therefore, conclude that the preoperational monitoring program for Units 2 and 3 is acceptable. The applicant's program has not yet been reviewed by the Fish and Wildlife Service.

2.7 Radioactive Waste Control

As with similar PWR plants, the radioactive waste handling storage equipment are sized on the basis of continued reactor operation with clad defects in one percent of the fuel rods. The applicant has stated that liquids are normally held for reuse or for shipment offsite with the optional capability of discharge to the circulating water outfall within the limits of 10 CFR Part 20. Liquid releases to the circulating water discharge are continuously monitored for high radiation. High radiation will automatically close a discharge valve in the liquid waste disposal system. Samples of liquid effluents will also be taken to insure conformance with the release limits. Solid radioactive wastes from plant operation will be temporarily stored onsite and shipped from the site in containers approved for that puspose. Gaseous wastes from the chemical and volume control system, various gases and vents will be collected and compressed into gas decay tanks. Six such tanks will be provided and after suitable decay period (a minimum of 30 days) the contents from a tank will be released through filters to the stack vent. A radiation monitor is provided in this discharge line and if the radiation level becomes high during a release, a valve in the discharge line from these tanks will close automatically. The applicant estimates that the gaseous activity released to the environment will be on a yearly average of 0.1 percent of 10 CFR Part 20 values. We have requested detailed information from the applicant on the design characteristics of the radwaste system with an indication of the quantity of each radionuclide released to the environment and a comparison of this release with the limits of 10 CFR Part 20. From the limited description of the waste disposal system contained in the PSAR, we tentatively conclude that this system is proposed by the applicant will provide effective control for radioactive waste generated at the site and that it is acceptable.

ACCIDENT ANALYSIS

GENERAL

We have postulated several principal situations leading to a major release of fission products to the environment. All calculated doses are within 10 CFR 100 guidelines, the most significant being the LOCA thyroid doses: 214 rem at the exclusion zone distance (804 meters) and 81 rem at the low population zone distance, (1.84 km or 3 miles). These doses were computed based upon a containment leak rate of 0.1% per day under assumed accident conditions.

TABLE 14.1

ACCIDENT CONSEQUENCES

ACCIDENT	TWO HOUR SITE BOUNDARY DOSES (REM)		LPZ COURSE OF ACCIDENT DOSES (REM)	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss of Coolant	214	3.42	81.0	1
Refueling	145	2.6	13.0	1
Gas Decay		13		1.0
Tank Rupture				

The staff standard meteorological assumptions for the short term releases are Pasquill Type "F" and $\bar{u}$ = 1 meter/second. The site boundary X/Q value using these assumptions is a factor of 3.9 higher than those that are calculated by the applicant using Pasquill Type "E" meteorology and a 2.2 meter per second wind speed. It is the staff's opinion at this time that the Δt measurements may influence the standard staff meteorological assumptions.

Due to the seacoast location of the site, both stability and wind conditions tend to make diffusion conditions more favorable than the standard staff model assumptions. This may result in a reduction of the doses by as much as a factor of three.

If the applicant's Δt meteorological program confirms diffusion conditions better than those now used by the staff, a containment building leak rate greater than 0.1% could be permitted in the technical specifications of the time the operating license for these plants is issued.

On the basis of our evaluation, we conclude that in the unlikely event of any of the postulated accidents, the resulting calculated radiological consequences would not exceed the guidelines of 10 CFR 100.

ACCIDENT ASSUMPTIONSS

A. Loss-of-Coolant Accident Assumptions

The loss-of-coolant accident dose analysis was based on the following assumptions:

1. Power level of 3390 MWt.
2. TID-14844 releases (100% noble gases, 50% of the iodines, 1% of the solids, 50% plateout of halogens).
3. Design containment leak rate of 0.1%/day for the first day and 0.05%/day thereafter.
4. Spray removal constant for inorganic iodines of 5.8 hours^{-1}.
5. Standard ground release meteorology and dose conversion factors. (Pasquill Type "F" and u = one meter/sec. for short term doses).

B. Refueling Accident Assumptions

The refueling accident dose analysis was based on the following assumptions:

1. Rupture of 176 fuel rods (one whole assembly).
2. Gap activity in the rods is released. This is assumed to be 20% of the noble gases and 10% of the iodine in the fuel rods, with a peaking factor of 1.8.
3. The accident occurs after 72 hours cooling time.
4. 90% of the iodine is retained in the pool water.
5. Standard ground release meteorology and dose conversion factors.
6. Iodine removal factor of 10 for filters.

C. Gas Decay Tank Rupture Assumptions

Our calculation of the potential doses resulting from the rupture of gas decay tank was based on the following assumptions:

1. Gas Decay tank contains one complete primary coolant loop

inventory of noble gases resulting from operation with 1% failed fuel (159,000 curies equivalent of Xe^{133}).

2. An average decay energy of 0.7 Mev for the fission product mixture.

3. Standard ground level release meteorology and dose conversion factors.

Suppl F.
Doc 50-361

APR 26 1971

P. A. Morris, Director, Division of Reactor Licensing

SOUTHERN CALIFORNIA EDISON COMPANY - SAN ONOFRE UNITS NO. 2 AND 3
DOCKET NOS. 50-361/362

Adequate responses to the enclosed request for additional information are required before we can complete our review of the subject application. These requests, prepared by the DRS Mechanical Engineering Branch, concern design of valves, design of reactor internals for the Design Basis Accident, and stress limits for faulted conditions. The enclosed supplements our requests for information of September 11, 1970, March 4 and 23, 1971 and April 12, 1971.

Original Signed By
E. G. Case

Edson G. Case, Director
Division of Reactor Standards

Enclosure:
Additional Information Requests
for San Onofre Units 2 and 3

cc w/encl:
S. H. Hanauer, DR
R. DeYoung, DRL
R. Boyd, DRL
D. Skovholt, DRL
R. R. Maccary, DRS
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R. Birkel, DRL
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J. Knight, DRS

OFFICE ▶ CRESS:tlc T31A R01-05	DRS:MEB	DRS:A/D	DRS:DIR			
SURNAME ▶	D. Lange K. Wichman:tlc	R. R. Maccary	E. G. Case			
DATE ▶ 4/23/71	4/23/71	4/23/71	4/24/71			

Memo 1

REQUEST FOR ADDITIONAL INFORMATION

SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1. Table 4.3-1 of the PSAR specified the ASME Code for Pumps and Valves for Nuclear Power (NP&V Code) as applicable to the design of Class 1 (NP&V Code) valves. However, this Code, in conjunction with two recent code case interpretations, allow the option of selecting any of the following design procedures for Class 1 valves:

   a. Paragraph 452.1a of the NP&V Code, Standard Pressure Rated Valve

   b. Paragraph 452.1b of the NP&V Code, Non-Standard Pressure Rated Valve

   c. MSS SP-66, Pressure-Temperature Ratings for Steel Butt-Welding End Valves, 1964 edition (as referenced in the March 1970 Addenda to the NP&V Code)

   d. ASME Code Case 1465

   e. ASME Code Case 1466

   Indicate which of the above designs standards will be used in the design of Class 1 valves within the reactor coolant pressure boundary.

2. The ASME Code for Pumps and Valves for Nuclear Power (NP&V) which is specified for the design of pumps and valves within the reactor coolant pressure boundary for this plant stipulates the use of ANSI B31.7 Nuclear Power Piping Code for design under earthquake loadings (paragraph 424). For the combination of loadings which include those due to earthquake, emergency and faulted operating condition categories may apply in conjunction with the associated stress limits as given in Case 70 of Interpretations of Code for Pressure Piping. Indicate whether the stress limits of Code Case 70 will be applied in the design of pumps and valves within the reactor coolant pressure boundary for the emergency and faulted operating condition categories. If other stress criteria are proposed, provide the basis for their application.

3. Distinguish between the stress limits proposed for active[1] and inactive[2] pumps and valves, e.g. certain pumps and valves

---

1 Inactive components are those whose operability (e.g., valve opening, or closure, pump operation or trip) are not relied upon to perform the system function during the transients or events considered in the respective operating condition categories.

2 Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered in the respective operating condition categories.

(classified as active components) within the reactor coolant pressure boundary and required not only to serve as a pressure-retaining component (as in the case of passive components, vessels, and piping) but also to operate reliably in order to perform a safety function; such as safe shutdown of the reactor, or, in the event of a pipe break in the system, to mitigate the consequences of the accident under the loading combinations considered in design. Therefore, to assure that an active component will function (e.g. closure of containment isolation valves) in the event of a pipe rupture in the reactor coolant pressure boundary (faulted condition), we consider the stress limits for the "emergency condition" as appropriate in lieu of the code stress limits for the "faulted condition". State whether it is your intention to comply with the limits indicated for active pumps and valves as defined, or justify any exceptions in your response.

4. For the combination of normal plus pipe rupture plus Design Basis Earthquake loadings the proposed primary stress limits applicable to vessels and piping within the reactor coolant pressure boundary exceed the component code stress limits considered appropriate for the faulted operating condition category (i.e. limit analysis of the ASME Section III Nuclear Vessel Code and Case 70 of

Interpretations of Code for Pressure Piping applicable to nuclear power piping). Document your intention to comply with the applicable code limits for "faulted conditions" or present justification for exceeding those limits. If you intend to use the "simplified" analysis of I-705 of the ANSI B31.7 Nuclear Power Piping Code for design of piping within the reactor coolant pressure boundary, confirm that the stress limits employed for emergency and faulted operating condition categories will not exceed the respective limits of Code Case 70.

6. Results from analyses and tests of reactor internals configurations similar to the internals of this plant have substantiated that the maximum responses of the core barrel under the forcing functions associated with the Loss-of-Coolant Accident (LOCA) have emanated from the circumferential or "ring modes" of vibration. Indicate whether the experimental or theoretical design analyses which will be performed for the core support barrel under LOCA conditions will consider the effects of the ring modes (n = 0, 2, 4, 6, etc.) and the lateral pressure maldistribution that occurs across the core support barrel during the LOCA. Submit the basis for your design approach.

DISTRIBUTION:
Docket No. 50-361
Docket No. 50-362
DRL Reading
PWR-3 Reading

APR 23 1971

R. C. DeYoung, Assistant Director for PWRs, DRL
THRU: K. R. Goller, Chief, PWR Branch No. 3, DRL

SOUTHERN CALIFORNIA EDISON COMPANY - SAN DIEGO GAS & ELECTRIC COMPANY
SAN ONOFRE 2/3 - DOCKET NOS. 50-361 & 50-362

Enclosed is a summary of the meetings held with Southern California Edison Company and San Diego Gas & Electric Company on April 6 & 7, 1971, in Bethesda, Maryland. An attendance list is also enclosed.

Original Signed by
R. A. Birkel

Ralph A. Birkel
PWR Branch No. 3
Division of Reactor Licensing

Enclosures:
1. Meeting Summary
2. Attendance List

cc w/encls:
P. A. Morris, DRL
F. Schroeder, DRL
T. R. Wilson, DRL (2)
R. S. Boyd, DRL
D. J. Skovholt, DRL
N. M. Brown, DRL
E. G. Case, DRS
DRL/DRS Branch Chiefs
R. W. Klecker, DRL
Meeting Attendees from REG.

OFFICE ▶	DRL: PWR-3 x-7243 RAB	DRL: PWR-3 KRG				
SURNAME ▶	RABirkel:tls	KRGoller				
DATE ▶	4/22/71	4/22/71				

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

Memo

ENCLOSURE NO. 1

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS & ELECTRIC COMPANY

SAN ONOFRE UNITS 2/3 - DOCKET NOS. 50-361 & 50-362

MEETING SUMMARY

Abstract

Technical meetings were held in Bethesda on April 6 & 7, 1971, principally to review nuclear design matters, primarily adequacy of the applicant's response to formal staff nuclear design questions. In addition, sundry remaining open review items were discussed including quality group classification, flywheel criteria, vibration testing, safety guides, control of combustible gas following LOCA, GDC - 1971, emergency plans and tornado design criteria.

April 6, 1971

Messers Dunenfeld and Richings lead the discussion on the proposed nuclear design for the San Onofre plants. The discussion revolved primarily around the staff evaluation of the applicant's response to the February 9, 1971, formal staff questions. The overall intent of the latter was to obtain more information concerning: (1) the designer's ability to obtain the low nuclear peaking factor upon which the power level of the power plant is dependent, (2) the adequacy of the instrumentation system to monitor, detect and provide protection from power maldistributions and (3) the adequacy of power maldistribution control provisions. The responses provided either evaded the issue or asserted that there were no problems, but provided no information to support such conclusions.

A total of 12 responses were discussed in detail with the applicant, Questions 3.29, 3.33, 3.34, 3.35, 3.36, 3.37, 3.38, 3.39, 3.40, 3.41, 3.42 and 14.10. Of these, one response (3.37) was found to be satisfactory based upon discussions presented by the applicant, two responses (3.35 and 3.39) would be revised to indicate an applicant commitment that automatic reactor protection would be provided unless it could be conclusively shown that fuel damage will not occurr and the remaining nine responses would be upgraded and revised.

April 7, 1971

In brief, the following topics were reviewed:

1. Quality Group Classification (question 4.1). The applicant provided the staff with a draft of his proposed quality group classification system including process diagrams suitably marked to indicate QGC. Since the proposed system is unique to this application, the staff suggested that the applicant consider using the Codes & Standards Supplement, October 20, 1970, as a guide in finalizing their QGC system. Copies of the supplement were provided to the applicant. (Maccary, Kirkwood, Arlotto)

2. Flywheel Criteria. The following criteria were discussed with the applicant. (Fairtile)

   a. Fracture Toughness

      1. Estabilsh NDTT = 10°F max by Drop Wt. test

      2. Establish Fracture Toughness Energy of 50 feet. pounds minimum at minimum operating temperature (in range of 100 - 110°F usually) by 3 Charpy Cv samples in each rolling direction.

   b. Material

      High quality such as vacuum melt and degassed - idea being to limit inclusions, slag and gas bubbles.

   c. Initial Inspection

      1. After flywheel fabrication 100% UT

      2. Surface, such as Mag. Particle or Dye penetrant of bore, keys and other machined surfaces.

   d. Inservice Inspection

      1. Either access for 100% UT, or

      2. Removability of flywheels for 100% UT.

   The applicant agreed to review these criteria.

3. Vibration Testing. The requirements for vibration testing for San Onofre 2/3 as stated in E. G. Case's memo dated March 23, 1971, was discussed with the applicant. (Lange, Wichman)

4. Safety Guides. The forthcoming safety guides numbers 5 through 13 were discussed. (Birkel)

5. Control of Combustible Gas Following LOCA. The applicant agreed to provide the staff with a design of their proposed system approximately 6 months after issuance of the CP. (Birkel)

6. General Design Criteria - February 1971. The applicant indicated that these criteria are being reviewed and informal comments will be provided to the staff. Combustion Engineering stated that their corporate comments would be mailed to the Commission later this month. (Birkel)

7. Exemption Request. The applicant stated that at this time there is no intention of filing an exemption request prior to issuance of a CP. The project schedule has slipped on a month for month basis since February 1, 1971. (Birkel)

8. Emergency Plans, Conduct of Operation (SRO's) and Quality Assurance Program. (McGough, Birkel)

   a. Emergency plans require revision to indicate clearly the boundaries for the restricted area, on-shore and possibly off-shore. In addition, the control the applicant has regarding evacuation of the public off-shore from the site must be clearly indicated. The applicant indicated that his emergency plan for Unit 1 was currently under review. SCE stated they would consider providing additional information in the PSAR for Units 2 and 3 regarding this area of concern.

   b. Conduct of Operations (Senior Reactor Operators). The staff indicated that the proposed <u>6</u> man shift operating crew for Twin reactors is below acceptable standards currently in use, however this aspect will be evaluated during the POL review. The proposed single SRO per shift operating crew is however unacceptable. A minimum of <u>2</u> SRO's will be required per shift operating crew. The applicant indicated that they could probably meet such a requirement if position titles would not dictate SRO requirements.

   c. Quality Assurance Program. Material traceability was inadequately described in the QAP. This aspect will be corrected by the applicant.

9. Tornado Design Criteria. This topic again aroused considerable discussion which is detailed in a separate memo to R. C. DeYoung. In summary, based upon the low probability of a Midwestern type design basis tornado occuring at the San Onofre site, the applicant does not consider it appropriate to apply the design basis tornado criteria to his design. In response to staff discussion the applicant has reviewed his plant design and has concluded that major modifications would be required to meet the design criteria. In addition, applying such criteria to Units 2/3 would have a deleterious effect on continued operation of Unit No. 1, especially if a contested ASLB hearing for Units 2/3 occurr. (Birkel)

ENCLOSURE NO. 2

SAN ONOFRE UNITS 2 & 3

DOCKET NOS. 50-361 & 50-362

MEETINGS ON APRIL 6 & 7, 1971

ATTENDANCE LIST

April 6, 1971

SOUTHERN CALIFORNIA EDISON CO.

L. D. Hamlin
K. P. Baskin

COMBUSTION ENGINEERING

W. R. Corcoran
R. W. Knapp
V. C. Hall
H. von Steiger

BECHTEL CORP.

A. H. Whitaker

USAEC/DRL

R. A. Birkel
D. Fisher

USAEC/DRS

M. Dunenfeld
H. Richings

April 7, 1971

SOUTHERN CALIFORNIA EDISON CO.

L. D. Hamlin
K. P. Baskin
H. B. Ray

COMBUSTION ENGINEERING

D. F. Streinz
V. C. Hall
H. von Steiger
D. R. Wade

BECHTEL CORP.

A. H. Whitaker
L. H. Curtis

USAEC/DRL

R. A. Birkel
*J. M. McGough

USAEC/DRS

*R. Maccary
*R. Kirkwood
*G. Arlotto
*D. Lange
*K. Wichman
*M. Fairtile

*Denotes Part Time

APR 17 1971

Peter A. Morris, Director, Division of Reactor Licensing

SAN ONOFRE NUCLEAR STATION, UNITS 2 AND 3, DOCKET NOS. 50-361/362

The information submitted by the applicant has been reviewed and evaluated by the Materials Engineering Branch, DRS. Their final evaluation of the information submitted to date, including Amendment 8, received March 8, 1971, is enclosed. Tentative conclusions are enclosed in parentheses; the summary of actions to be taken to resolve open issues is enclosed in brackets.

Original signed by
E. G. Case
Edson G. Case, Director
Division of Reactor Standards

Dkt 50/361

Enclosure:
Materials Engineering CP
Evaluation for San Onofre 2/3

cc w/encl:
S. H. Hanauer, DR
R. Boyd, DRL
R. DeYoung, DRL
D. Skovholt, DRL
R. Maccary, DRS
K. Goller, DRL
R. Birkel, DRL
S. Pawlicki, DRS
L. Porse, DRS
M. FAirtile, DRS
R. Ferguson, DRS
A. Dromerick, DRS
D. Lange, DRS

CRESS
T-55:R-1:en

OFFICE ▶	DRS	DRS	DRS	DRS	DRS	
SURNAME ▶	MFairtile MBF	LPorse	SPawlicki	RMaccary	EGCase	
DATE ▶	4/15/71	4/15/71	4/16/71	4/16/71	4/16/71	

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2&3
DOCKET NOS. 50-361/362
FINAL CP REPORT - MATERIALS ENGINEERING BRANCH, DRS

Section numbers and headings correspond to interim report dated November 20, 1970.

4.0 REACTOR COOLANT SYSTEM

4.3 Missile Protection and Flywheel Integrity

4.5 Electroslag Welding

4.6 Reactor Coolant System Sensitized Stainless Steel

4.7 Foreign Procurement

4.9 Inservice Inspection

4.11 Reactor Coolant System - Fracture Toughness

4.3 Missile Protection and Flywheel Integrity

The design criteria presented in the PSAR, and expanded on in answer to our questions that govern the design of the missile protection features of the San Onofre 2/3 Station, are adequate in regard to internal missiles. Internal missile protection will be provided by separation of redundant engineered safety system equipment, use of missile shielding, proper equipment orientation, and by the consideration of design of potential missile sources. The applicant has documented both the concrete and the steel missile penetration formulas and applicable analytical methods which are acceptable.

We conclude that the design criteria proposed by the applicant will provide a basis for implementing adequate missile protection to the San Onofre 2/3 containment which will be reviewed at FSAR stage.

The primary system pump flywheels in the San Onofre Units 2 and 3 will be identical to those used in other plants. The flywheels will be of Allis-Chalmers design, fabricated of vacuum melt and degassed steel. The finished flywheels will be subjected to 100 percent volumetric UT inspection. Finished machined bores will also be subjected to magnetic particle or liquid penetrant examination.

The applicant has stated that a minimum of three Charpy V-notch (Cv) specimens will be tested from each plate, parallel and normal to the rolling direction, to determine fracture toughness of the material. A Nil-Ductility Transition (NDT) temperature of 10°F maximum will be specified for the flywheel material.

[We believe that the proposed NDT temperature will assure adequate fracture toughness of the flywheel material, provided that the NDT temperature is obtained from the dropweight tests, rather than Cv tests. We recommend also that the Cv tests be performed at the minimum operating temperature of the flywheels to establish fracture toughness in the vicinity of the Cv upper shelf energy.]

The applicant has stated that the pump flywheels will be accessible for inservice inspection.

(We conclude that the proposed design and fabrication procedures will assure adequate integrity of the pump flywheels at the San Onofre Units 2 and 3 and that the access provisions for inservice inspection will enable periodic verification of the flywheel integrity during their service life.)

4.5 Electroslag Welding

The applicant has stated that there are no plans to accept the use of electroslag welding process for any of the reactor coolant pressure boundary components. However, components such as pump bodies which have not yet been purchased might require electroslag welding. We expect to receive confirmation from the applicant if such welding will be used in pump bodies.

4.6 Reactor Coolant System Sensitized Stainless Steel

The applicant has stated that sensitization of non-stabilized stainless steel will not be permitted in the construction of the reactor coolant system and safety related parts. The precautions which will be employed to ensure this include control of the manufacturing sequence to eliminate heat treatments or any other process that might produce sensitization. The application of reactor vessel nozzle safe-ends is eliminated since the main coolant piping, except for the pressurizer surge pipe, is low alloy ferritic steel with stainless steel cladding on the inside, instead of the generally used stainless steel piping. The applicant has stated that any sensitized core structural components would pass a Strauss test. This test provides an acceptable basis for checking the presence of concernable effects of sensitization.

We conclude that the planning to avoid severe sensitization of austenitic stainless steel during the fabrication period is acceptable. The applicant has been requested to formally submit information regarding the use of nitrogen bearing stainless steel in any component within the reactor coolant pressure boundary, since such compositions may introduce some susceptibility to sensitization. The applicant has orally informed us that he will specify non-nitrogen bearing S.S.

4.7 <u>Foreign Procurement</u>

The applicant has stated that no major Class I components will be designed and/or fabricated in a foregin country. Although the reactor vessel, pressurizer, steam generators, piping and pumps will be purchased domestically, the applicant will advise us if other Class I components, such as valves should be produced outside the USA.

4.9 Inservice Inspection

The applicant has stated he will provide access to the reactor coolant pressure boundary in compliance with Section XI of the ASME Code. For engineered safety systems beyond the limits of the reactor coolant pressure boundary, which are not presently covered by Section XI of the ASME Code, the applicant will provide access to pipe welds, and pump and valve bodies in these systems. We conclude that the access provisions for the inservice inspection program are adequate.

4.11 Reactor Coolant System - Fracture Toughness

The following components of the reactor coolant pressure boundary which are made of carbon or low-alloy steel are required to meet fracture toughness properties in order to provide protection against brittle fracture under normal reactor operating condition.

Reactor Vessel	-	Shell	SA-533 Grade B, Class 1
Reactor Vessel	-	Forgings	A-508, Class 2
Pressurizer	-	Shell	SA-533 Grade B, Class 1
Steam Generator	-	Shell	SA-533 Grade A and B, Class 1
Steam Generator	-	Head	SA-508, Class 2
Reactor Coolant Piping			SA-516 Grade 70

The impact properties of all carbon steel and low-alloy steel, which form a part of the pressure boundary of the reactor coolant system, will comply with the requirements of the ASME Code, Section III, Paragraph N-330, at 40°F.

The reactor vessel will be designed to meet the impact test requirements specified in the rules of the ASME Boiler and Pressure Vessel Code, Section III. In addition, the pre-irradiated NDT temperature of the reactor vessel materials will be established using drop-weight tests, and correlations will be made with Charpy impact specimens test data.

For reactor operation at 3500 Mwt, and an 80 percent load factor, the neutron fluence ($E \geq 1$ Mev) at the inner surface of the vessel has been calculated by the applicant not to exceed 3.7 x 10^{19} nvt. Using the "worst case" curve (Fig. 4.3-1 of the PSAR) the maximum predicted increase in the NDT temperature is 265°F. The applicant has proposed to base the operating conditions of the plant on the results of the reactor vessel material surveillance program and on maintaining a 60°F temperature margin above the NDT temperature which is in compliance with ASME Code rules. However, recent fracture toughness test data indicate that the current ASME Code rules are not always sufficiently conservative, and may not guarantee adequate fracture toughness of ferritic materials. Quite often considerable difficulty exists in defining the transition temperature region from Charpy V-notch test data. In addition, the transition temperature itself depends on the thickness of the specimen tested (size effect).

[The applicant has been requested to indicate the extent to which his design criteria will meet the proposed Fracture Toughness Criteria, 10 CFR Part 50.55a(i), Appendix F.

In addition, in order to complete our evaluation of the adequacy of the fracture toughness of all pressure-retaining ferritic components of the reactor coolant pressure boundary, we have requested the applicant to provide us with the following:

(a) Fracture toughness data (Charpy V-notch fracture energy curves and Drop-Weight Test NDT temperature) for plates, forgings, piping, and weld metal,

(b) The proposed heatup and cooldown curves which will control the pressure and temperatures to which the material of the reactor coolant pressure boundary will be exposed at the end of its service life.

For reactor vessel beltline materials, including welds, we have also asked for the highest predicted end-of-life transition temperature corresponding to the 50 ft-lb value of the Charpy V-notch fracture energy for weak direction, and for the minimum upper shelf energy value for weak direction, which will be acceptable for continued operation toward the end of service life of the vessel.]

When sufficient information is made available, we will apply the AEC fracture toughness criteria to establish appropriate heatup and cooldown limits for this plant.

APR 1 2 1971

P. A. Morris, Director, Division of Reactor Licensing

SOUTHERN CALIFORNIA EDISON COMPANY - SAN ONOFRE UNITS NO. 2 & 3 DOCKET NOS. 50-361/362

An adequate response to the enclosed request for additional information prepared by the Division of Reactor Standards will be required before we can complete our review of the subject application. This request is in addition to the information requests submitted to DRL on November 18, 1970 and March 23, 1971.

Original signed by
E. G. Case

Edson G. Case, Director
Division of Reactor Standards

Dkt 50-361

Enclosure:
Additional Information Requests

cc w/encl:
S. H. Hanauer, DR
R. DeYoung, DRL
R. Boyd, DRL
D. Skovholt, DRL
R. R. Maccary, DRS
K. Goller, DRL
D. Lange, DRS
R. Birkel, DRL
K. Wichman, DRS

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OFFICE ▶	MEB:DRS	A/D:DRS	DIR:DRS			
SURNAME ▶	Lange Wichman:dlp	Maccary	EGCase			
DATE ▶	4/8/71	4/9/71	4/12/71			

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memo

Additional Information Request
San Onofre Units 2 & 3
Docket Nos. 50-361/362

Paragraph I-701.5.4 of the ANSI B31.7 Nuclear Power Piping Code requires that piping shall be supported to minimize vibration and that the designer is responsible by observation under startup or initial operating conditions to assure that vibration is within acceptable levels. Submit a discussion of your vibration operational test program which will be used to verify that the piping and piping restraints within the reactor coolant pressure boundary have been designed to withstand dynamic effects due to valve closures, pump trips, etc. Provide a list of the transient conditions and the associated actions (pump trips, valve actuations, etc.) that will be used in the vibration operational test program to verify the integrity of the system. Include those transients introduced in systems other than the reactor coolant pressure boundary that will result in significant vibration response of reactor coolant pressure boundary systems and components.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Docket 50-361

APR 9 1971

Original Signed By

R. C. DeYoung, Assistant Director for PWR's, DRL K. R. Goller
THRU: K. R. Goller, Chief, PWR Branch No. 3, DRL

AEC TORNADO DESIGN CRITERIA AS APPLIED TO THE SAN ONOFRE UNITS 2 AND 3 - DOCKET NOS. 50-361 AND 50-362

The staff discussed with the applicant soon after receipt of the CP application the extent to which the AEC tornado design criteria will be applied in the design of the San Onofre 2/3 plants. The staff's concern was expressed in a formal staff question (2.4) contained in our letter to the applicant dated October 28, 1970. Subsequently, this matter was again discussed with the applicant during the ACRS Subcommittee meeting at the site in January 1971.

The applicants position of not applying the AEC tornado design criteria to San Onofre 2/3 is based primarily on the lower probability of a tornado occurring at the San Onofre site which has been stated as 1-2 orders of magnitude lower than a typical Midwestern location. In addition, the applicant maintains that California tornadoes are significantly less severe than the Midwestern type. On the basis of the past 17 years, the applicant estimates that only about 1 in 10 California tornadoes might exceed a speed of 100 mph. If a conservative view is taken that only 1 of 5 tornadoes would exceed 100 mph, then the return period at the San Onofre site could be considered to be 139,000 years for a tornado to occur at that point that would have associated winds in excess of 100 mph. Based on the foregoing the applicant does not consider that any design basis tornado should be established for the San Onofre site.

Discussion with the applicant has revealed that extensive plant design changes would be required to meet the AEC design basis tornado. Initially it was felt that the most vulnerable areas were the roof of the control room and the roof of the auxiliary building (spent fuel pool area) however, subsequent review by the applicant has shown that extensive modifications would also be required of other safety related systems and equipment. Information obtained from the applicant on a verbal basis regarding these modifications is provided in the attachment.

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memo

The applicant further indicated that application of the Midwestern design tornado criteria to San Onofre 2/3 could have deleterious effects on the continued operation of Unit 1 with respect to intervention during the ASLB hearings.

Management guidance is requested regarding the interpretation and extent to which DRL will require San Onofre 2/3 to provide a design to meet the staff design basis tornado criteria.

Original Signed By
K. R. Goller KRG

Ralph A. Birkel
PWR Branch No. 3
Division of Reactor Licensing

Enclosure:
Effect of AEC Design Basis Tornado Criteria on San Onofre 2/3

cc: R. S. Boyd, DRL
D. R. Muller, DRL
C. G. Long, DRL
R. Klecker, DRL
N. M. Brown, DRL
Attorney, OGC

DISTRIBUTION:
Docket 50-361
Docket 50-362
DRL Reading
PWR-3 Reading
RABirkel, DRL

OFFICE ▶	DRL:PWR-3 x7415	DRL:PWR-3 KRG				
SURNAME ▶	RABirkel:esp	KRGoller				
DATE ▶	4/ /71	4/9/71				

EFFECT OF AEC DESIGN BASIS TORNADO

CRITERIA ON SAN ONOFRE UNITS 2 AND 3

The following systems and equipment may be considered to be vulnerable to isolated damage with respect to design basis tornado induced wind, vacuum and missile generated loads. Damage may be considered as originating from local or general failure of steel or concrete enclosures either surrounding the particular system or equipment or located adjacent to it:

1. Main steam lines and associated valves from the containment walls to the turbine.
2. Main feedwater and auxiliary feedwater lines and associated valves.
3. Containment ventilation components from the containment boundary.
4. Primary plant make up tanks.
5. Diesel generator system.
6. Elements of the process instrumentation and controls.
7. Emergency power supply systems and related electrical penetrations.
8. Salt water cooling pumps.
9. Concentrated boric acid storage tank.
10. Primary and secondary radwaste storage tanks.

A review of the general design modifications which would be required to withstand the major loading conditions consistent with the requirements of the design basis tornado are indicated below.

A. Structures

1. Containment Structure

No major modifications would be required.

2. Turbine Building Structure

Modifications for the turbine building would include:

(a) Tie-down of the turbine gantry crane with modifications of the crane supporting system in the absence of detailed analytical assurance that overturning is not feasible.

(b) Anchorage of major supporting sub-walls and floors to preclude effects of reversal of gravity loadings which may cause secondary structural damage to adjacent Class I or II facilities.

(c) Replacement of reinforced concrete for all masonry retaining walls to preclude secondary failure of critical facilities due to reinforced masonry failures.

(d) Special design requirements for positive tie-down of the turbine shelter structure to preclude secondary damage throughout the plant and damage or short-circuiting of the main take-off towers.

(e) Re-design of anchorage systems for Class III vessels which may cause secondary damage to Class I or II systems and equipment.

(f) Major modifications to segments of structural steel supporting systems currently governed by 100 year wind conditions (to the extent necessary to preclude progressive failure of the turbine support system) as related to the protection of adjacent Class I or II structures, systems, and equipment.

3. <u>Circulating Water Intake Structure</u>

No major modifications would be required.

4. <u>Auxiliary Building and Adjacent Structures</u>

Modifications to the auxiliary building would include the following:

(a) Replacement of thick reinforced concrete walls and roofing systems for combined steel beam, steel decking, and masonry over the spent fuel storage pool. Slab thicknesses would vary between 1'-0 and 2'-0 minimums with reinforced concrete supporting beams as large as 5'-0 in depth. Reinforcing steel quantities would be nearly double those of a non-tornado resistant structure.

(b) Incorporation of fuel handling and cask crane interlocks to assure positive stability for portions exposed to tornado conditions.

(c) Major modification of reinforced concrete walls and roofing system for the control room area. Wall thicknesses currently required for shielding would require additional reinforcing steel increases of nearly 100 percent to withstand suction loadings coupled with load reversals due to wind thrusts. Beam sections would require depths as large as 5'-0" with prestressing employed due to large spans involved. The use of structural steel and masonry would be precluded.

(d) All miscellaneous auxiliary building related trenches and sub-buildings would require substantial increases in reinforcing to assure that secondary failures of these structures would not damage Class I or II items.

B. Systems and Equipment

Protection of all Quality I or II and Seismic Class I systems and equipment located adjacent to major buildings or running between major buildings would require a minimum reinforced concrete enclosure of 1'-0 to preclude penetration and subsequent damage from tornado induced missile. Many portions of these systems and equipment currently are not specifically enclosed. In addition, the enclosures would have to be designed to withstand the postulated 300 mph wind loads and checked for bursting pressures of 3 psi where geometrical configuration dictate.

The design of enclosures would be required around all Class I or II tanks and vessels not capable of withstanding major wind loads; tornado vacuum loads; or missile loadings without major structural modifications. As such, protective walls would be required to ensure continued integrity based upon the function of the tank. Special modifications of Class III tanks and vessels would not result in residual damage to major Class I or II components.

APR 1 1971

Peter A. Morris, Director
Division of Reactor Licensing

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3
DOCKET NOS. 50-361/362

The additional information submitted by the subject applicant with respect to the containment and Class I structural design has been reviewed and evaluated by the DRS Structural Engineering Branch. The material reviewed to date is through Amendment No. 8 dated March 4, 1971. An evaluation of the information submitted to date is attached hereto as revisions to the interim evaluation review submitted to you by memorandum dated November 12, 1970.

Original signed by
E. G. Case

Edson G. Case, Director
Division of Reactor Standards

Enclosure:
Final ACRS Report
San Onofre, Units 2 and 3

cc w/encl:
S. H. Hanauer, DR
R. Boyd, DRL
R. DeYoung, DRL
D. Skovholt, DRL
R. Maccary, DRS
K. Goller, DRL
A. Dromerick, DRS
R. Birkel, DRL
A. Gluckmann, DRS
R. Shewmaker, DRS
G. Arndt, DRS
F. Schauer, DRS

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SURNAME ▶	RShewmaker:cn	ADromerick	RMaccary	EGCase		
DATE ▶	3/29/71	3/30/71	3/31/71	3/31/71		

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Memo

SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 AND 3

Docket Nos. 50-361/362

DRS FINAL EVALUATION REVIEW

Revisions of Interim Evaluation Review

(Dated November 12, 1970)

1. Revise the third paragraph, enclosed in brackets, under the section entitled, Class I Structures, to read as follows and remove the brackets:

   "The applicant's program of quality control for the Cadweld splices meets the AEC Guide on Cadweld splices and is judged to be an adequate program."

2. Revise the last paragraph, enclosed in brackets, under the section entitled, Foundation and Environmental Considerations, to read as follows and remove the brackets:

   "The applicant has stated that Units 2 and 3 will have additional seismic recording equipment in addition to that used in Unit 1. The installation will be similar to the Unit 1 system, the MTS-100. The staff plans to require a system equivalent to that of the AEC Safety Guide on Instrumentation for Earthquakes."

3. Revise the second paragraph under the section entitled, Containment Description, Design Criteria, and Loads by the deletion of the brackets and the two sentences at the end which are in the brackets.

4. Revise the section entitled, Testing and Surveillance, by removing the sentences within the brackets and the brackets and adding the following.

> "Bechtel Corporation has submitted a report to justify a tendon surveillance program. It is the intent of the staff to apply the resolved and accepted program to all Bechtel designed containments."