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MEMORANDUM FOR: F. G. Pagano, Chief, Emergency Preparedness Licensing Branch, DEP
T. P. Speis, Chief, Reactor Systems Branch, DSI

FROM: F. J. Miraglia, Chief, Licensing Branch No. 3, DL

SUBJECT: San Onofre 2 & 3, Low Power License

Southern California Edison (SCE) has filed a motion with the ASLB for consideration of issuance of a low power license for San Onofre 2 and 3.

As a result, it will be necessary for Emergency Preparedness staff and Reactor Systems staff to prepare testimony similar to that filed in the Diablo Canyon proceeding.

My understanding is that this testimony is to be filed by September 18, 1981. The case attorneys have requested that draft testimony be provided to them by c.o.b. September 11, 1981.

I have enclosed a copy of the Diablo Canyon testimony for your use (Note: Testimony on block valves is not relevant to the San Onofre proceeding).

It is imperative that the above schedule be met so as not to delay the San Onofre 2 & 3 proceeding. Please inform me (X29779) or Harry Rood (X28544) of the names of the individuals who will prepare the requested testimony, no later than c.o.b. September 4, 1981. The project manager and case attorneys will meet with these individuals at an early date.

Original signed by
Frank J. Miraglia

Frank J. Miraglia, Chief
Licensing Branch No. 3
Division of Licensing

cc:w/o encl:
D. Eisenhut
R. Tedesco
L. Chandler, OELD
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B. Buckley 144

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

May 8, 1981

John F. Wolf, Esq., Chairman
Administrative Judge
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Jerry Kline
Administrative Judge
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Glenn O. Bright, Esq.
Administrative Judge
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of
Pacific Gas and Electric Company
Diablo Canyon Nuclear Power Plant Unit Nos. 1 and 2
Docket Nos. 50-275 & 50- 323 O.L.

Dear Administrative Judge:

Enclosed find the NRC Staff Testimony of Bartholomew C. Buckley, John Sears, Frank C. Cherny, and Norman Lauben for the Diablo Canyon Low Power test proceeding. This testimony is filed pursuant to the schedule adopted by the Board in the February 13, 1981 Prehearing Conference Order.

Sincerely,

William Olmstead

for Bradley W. Jones
Counsel for NRC Staff

Enclosure:
As stated above

cc: Service List

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)
)
PACIFIC GAS AND ELECTRIC COMPANY) Docket Nos. 50-275 O.L.
) 50-323 O.L.
(Diablo Canyon Nuclear Power Plant)
Unit Nos. 1 and 2))

TESTIMONY OF BARTHOLOMEW C. BUCKLEY

Q. Please state your full name.

A. Bartholomew Christopher Buckley.

Q. By whom are you employed?

A. The Nuclear Regulatory Commission (NRC).

Q. Describe the nature of your work with respect to the Diablo Canyon operating license proceeding.

A. Since March 11, 1979, as a Senior Licensing Project Manager, I am responsible for the management of the NRC Staff's safety review and the preparation of safety evaluation reports in support of the issuance of operating licenses for Diablo Canyon, Units 1 & 2.

Q. Have you, in the course of your professional experience, had occasion to act as a project manager for plants other than Diablo Canyon?

A. Yes, the Rancho Seco and Crystal River Nuclear Power Plants.

Q. Would you detail your professional qualifications?

A. Attached is a copy of my professional qualifications.

Q. Are you familiar with the SER and supplements thereto which have been issued for the Diablo Canyon Nuclear Facility?

A. Yes.

Q. Did you participate in the preparation of those documents?

A. Yes.

Q. Please detail the extent of your participation in the preparation of the SER and Supplements thereto for the Diablo Canyon Nuclear Facility.

A. After receipt of SER inputs from the various technical branches, I prepared Supplements Number 9, 10, 11, 12, 13 and 14. The Safety Evaluation Report, dated October 16, 1974, and Supplements 1 through 8 were prepared by other project managers.

Q. Would you please identify all the individuals who prepared the sections of the SER and Supplements thereto which relate to the contentions admitted in this low power proceeding?

A. Emergency Planning: Peter Tam (former NRC employee); Relief and Safety Valves: Edgar Hemminger

Q. Have you reviewed the input to the SER of the above identified individuals?

A. Yes.

Q. Do you agree with the analysis and conclusions reached by those individuals within the limits of your professional expertise?

A. Yes.

Q. Considering that review do you, based on your professional experience, have any concern that the issues raised by the admitted contentions indicate any danger to the public safety and health by operation of the Diablo Canyon Nuclear Facility at low power?

A. No.

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	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant Unit Nos. 1 and 2)	)	

BARTHOLOMEW C. BUCKLEY

PROFESSIONAL QUALIFICATIONS

DIVISION OF LICENSING

I am a Senior Project Manager of the Staff of the U. S. Nuclear Regulatory Commission. I am responsible for the management of the NRC Staff's safety review and the preparation of safety evaluation reports in support of the issuance of operating licenses for Diablo Canyon, Units 1 and 2.

I received a Bachelor of Chemical Engineering Degree at the Catholic University of America in 1953 and I studied for a Master's Degree in Nuclear Engineering at the above cited University.

I started my employment with the Regulatory staff on January 3, 1972 and have worked as a Project Manager in Operating Reactors Branch and as a Section Leader in the Division of Operating Reactors.

I was employed at the Naval Nuclear Power Unit, Fort Belvoir, Virginia from 1966 to December 1971. I worked as a chemical engineer for approximately two years, and as Director of the Engineering Department until December 1971.

From 1960 to 1966 I worked at the Atomic Energy Division of Allis Chalmers. I prepared primary system and auxiliary preoperational test procedures and conducted and evaluated these preoperational tests at the Elk River Reactor. I also prepared a water chemistry manual and performed water chemistry analyses at the Elk River Reactor. I prepared all fluid systems preoperational test procedures for the LaCrosse Reactor and taught plant operating personnel in the design and operating aspects of certain systems.

I worked as a process control engineer at the MELPAR Co. located in Virginia from 1958 to 1960.

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	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant	)	
Unit Nos. 1 and 2)	)	

TESTIMONY OF JOHN R. SEARS

Q. Please state your full name.

A. John R. Sears.

Q. By whom are you employed?

A. I am employed by the U.S. Nuclear Regulatory Commission. I am a Senior Reactor Safety Engineer in the Emergency Preparedness Licensing Branch, Division of Emergency Preparedness of the Office of Inspection & Enforcement.

Q. Describe the nature of your work with respect to the Diablo Canyon operating license proceeding?

A. I am responsible for review and evaluation of the Emergency Plan for the Diablo Canyon reactors.

Q. Have you reviewed the emergency plans for plants other than Diablo Canyon?

A. Yes.

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Q. What previous review experience have you had with respect to emergency plans?

A. Prior to the summer of 1979, I was responsible for review of the emergency plans for all operating reactors.

Q. Would you detail your professional qualifications?

A. Attached is a copy of my professional qualifications.

Q. Has PG&E revised the Diablo Canyon Power Plant Emergency Plan?

A. PG&E filed a revision to the Diablo Canyon Power Plant Emergency Plan in February of 1980. PG&E also provided additional information in letters, PG&E to NRC, dated January 13, 1981 and February 27, 1981.

Q. Did you participate in the NRC review of these documents?

A. Yes.

Q. How was the Emergency Plan evaluated?

A. The Emergency Plan was reviewed against the specific criteria of the sixteen Planning Standards in Part II of the "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0654, Rev. 1, November 1980. The same sixteen standards are listed as requirements of 10 C.F.R. 50.47, Final Regulations on Emergency Planning, August 18, 1980. The NRC Staff has prepared an Emergency Preparedness Evaluation Report which lists each of the sixteen standards listed in 10 C.F.R. 50.47 in order, followed by a summary of applicable portions of the Emergency Plan as they apply to the Standard.

In addition, the Emergency Plan has been evaluated against the requirements of Sections II.A.1.1 and III.A.1.2 of NUREG-0694 (now superceded by the same sections of NUREG-0737).

Furthermore, California nuclear plant licensees and applicants have been requested to provide analyses on the effects of earthquake on their emergency plans, specifically in terms of the utilities' capabilities to insure availability of personnel and equipment to the sites.

Q. Where are the results and conclusions of the NRC Staff's evaluation reported?

A. The NRC Staff's results and conclusions are reported in Appendix B Emergency Preparedness Evaluation Report, to Supplement No. 14 to the Safety Evaluation Report, April 1981.

Q. What is the NRC staff conclusion?

A. The NRC Staff conclusion is that the Diablo Canyon emergency plan, when revised in accordance with the commitments made, provides an adequate planning basis for an acceptable state of emergency preparedness and will meet the requirements of 10 C.F.R. 50 and Appendix E.

Q. Do you concur in the results and conclusions of the Emergency Preparedness Evaluation Report?

A. Yes.

Q. Are there deficiencies in the Diablo Canyon Emergency Plan?

A. Yes.

Q. Has PG&E made commitments to correct these deficiencies?

A. Yes, PG&E has committed to correct these deficiencies before a license for full power is granted.

Q. What are these deficiencies?

A. The principal deficiency is that a fast public alerting system is not in place. 10 C.F.R. 50 Appendix E requires a demonstration of a fast alerting system to the general public. PG&E has described its proposed system in Section 7.2.11 of the Emergency Plan. PG&E has contracted for the installation of 52 electro-mechnical sirens to perform this function. Installation will commence as soon as necessary local permits are obtained.

Q. What is the significance of this deficiency for low power operation?

A. The requirement for a system to alert the public within 15 minutes is based on an accident scenario which could result in an offsite release of radiation in as little as thirty minutes from the time of initiation of an accident. The requirement is not necessary for operation at 5% power because, even if a credible LOCA occured, the resultant heat up of the core to melting point would take at least 15 hours and consequently a fast alerting system is not required.

Q. Is there a deficiency regarding a public information program?

A. Yes, 10 C.F.R. 50 Appendix E requires the dissemination to the public within the 10 mile Emergency Planning Zone of basic planning information. PG&E has a public information brochure in draft form. The final version must be integrated with the emergency plans of local governments which are presently being revised and upgraded.

Q. What is the significance of this deficiency for operation to 5%?

A. The purpose of the requirement for a public information program is to inform the public of how they will be alerted and what to do in an emergency. Since the alerting system is not yet in place, there is not yet a need for a public information program explaining the system.

Q. Are there other deficiencies in the Diablo Canyon Emergency Plan?

A. There are othe areas where the On-Site Plan is not in full compliance with the elements of the guidance in NUREG-0654. Many have been addressed in the letter, PGE to NRC, dated January 13, 1931.

Q. What is the significance of these deficiencies for 5% power?

A. These deficiencies are not significant for operation at 5% power. PG&E has committed in its letter to NRC, dated 2/27/81 that a full power license NUREG-0654 will be substantially complied with.

Q. Will a supplement to the Emergency Preparedness Evaluation Report be issued?

A. Yes, a supplement to the Emergency Preparedness Evaluation Report will be issued.

Q. What will the supplement contain?

A. The supplement will set forth the findings and determinations of the Federal Emergency Management Agency (FEMA) as to whether State and local emergency response plans are adequate and capable of being implemented for full power operation.

Q. Has FEMA made a finding for low power operation at Diablo Canyon Power Plant?

A. Yes.

Q. What was FEMA's finding for the adequacy of the present emergency plans at Diablo Canyon?

A. FEMA has made a finding that the present emergency plans at Diablo Canyon adequately protect the public health and safety for the purposes of low power testing.

Q. Where is FEMA's finding documented?

A. FEMA's finding is documented in a Memorandum for Harold R. Denton and John W. McConnel from FEMA/NRC Steering Committee and in the FEMA/NRC Interim Agreement on Criteria for Low Power Testing at New Commercial Nuclear Facilities -- SER Supplements Nos. 10 and 12.

Q. How is PG&E complying with the NRC request that all California nuclear plant licensees' and applicants provide analyses on the effects of earthquake on their emergency plans, specifically in terms of the utilities' capabilities to insure availability of personnel and equipment to the sites?

A. I have been informed by PG&E that a contractor, has been employed to study the impact of earthquakes on the emergency plans of the utility and of offsite authorities.

Q. When will the contractor issue their report?

A. The report is due in mid-May of 1981.

Q. How has PG&E proposed to utilize the contractor's report?

A. PG&E will revise the Emergency Plans to include the contractor's recommendations. This will be done prior to full power operation.

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Q. Are there presently emergency plans for the Diablo Canyon facility?

A. Yes.

Q. Does the licensee's plan presently have a notification requirement as regards accidents at the facility?

A. Yes.

Q. What does the licensee's present emergency plan require?

A. The licensee's present emergency plan requires prompt notification of the county governments surrounding Diablo Canyon Nuclear Facility. Governor Brown has admitted this.

Q. Did Governor Brown admit anything else as regards emergency planning?

A. Yes, both Governor Brown and Joint Intervenors have admitted that any accident occuring during low power testing would release a fraction of the existing fission product inventory at the Diablo Canyon Nuclear Facility.

Q. Are the present emergency plans in full force and effect?

A. Yes.

Q. What is your opinion about the adequacy of Diablo Canyon's combined Applicant, State and local emergency response plans if an accident were to occur during low power operation?

A. My opinion is that, if, during low power operation, an accident were to occur, the combined applicant, State and local emergency response plans will insure that an adequate state of emergency preparedness is in place to minimize the risks of hazard to the health and safety of the public during fuel loading and operation to 5% power.

UNITED STATES OF AMERICA
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BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant Unit Nos. 1 and 2)	)	

JOHN R. SEARS

PROFESSIONAL QUALIFICATIONS

EMERGENCY PREPAREDNESS PROGRAM OFFICE

Prior to 1952, I was employed in field jobs in various aspects of mechanical engineering. In 1952, I joined Brookhaven National Laboratory as a Reactor Shift Supervisor on the Brookhaven Graphite Reactor. While at Brookhaven, I completed a series of courses given by the Nuclear Engineering Department in nuclear engineering. These courses were patterned on the ORSORT programs. In 1956, I was appointed Project Engineer on the Brookhaven Medical Research Reactor. I was a member of the design group, participated in critical design experiments, wrote specifications, coauthored the hazards report, was responsible for field inspection and contractor liaison, trained operators and loaded and started up the reactor. About three months after start-up, in 1959, following the successful completion of proof tests and demonstration of the reactor in its design operating mode for boron capture therapy of brain cancer, I accepted a position as reactor inspector with the Division of Inspection, U. S. Atomic Energy Commission. In 1960, I transferred, as a reactor inspector, to the newly-formed Division of Compliance. I was responsible for the inspection, for safety and compliance with license requirements, of the licensed reactors and the fuel fabrication and fuel processing plants, which use more than critical amounts of special nuclear material, in the Eastern United States.

In September 1968, I transferred to the Operational Safety Branch, Directorate of Licensing. My responsibility included development of appropriate guides for evaluation of operationsl aspect of license applications and staff assistance in review of power reactor applicants submittals in the areas of Organization and Management. Personnel Qualifications, Training Programs, Procedures and Administrative Control, Review and Audit, Start-up Testing Programs Industrial Security and Emergency Planning.

The Branch was reorganized as the Industrial Security and Emergency Planning Branch in April 1974 to place increased emphasis and attention upon areas of physical security and emergency planning.

In 1976, I transferred to the Division of Operating Reactors as the sole reviewer responsible for review of emergency planning for all the operating reactors in the United States.

New York City College, 1950 - Mechanical Engineering

Argonne International School of Reactor Technology, 1961 - Reactor Control Course

GE BWR System Design Course, 1972

Popo-U.S. Army, 1974 - Course in Industrial Defense and Disaster Planning

Instructor at DCPA, 1976, 1977 - Course in Emergency Planning

Director, 1962 - Reactor Program, Atoms for Peace Exhibit, Bangkok, Thailand

Director, 1966 - Atoms for Peace Exhibit, Utrecht, Holland

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(Diablo Canyon Nuclear Power Plant	)	
Unit Nos. 1 and 2)	)	

TESTIMONY OF FRANK C. CHERNY
ON RELIEF, SAFETY AND BLOCK VALVES
AT DIABLO CANYON NUCLEAR FACILITY

Q. Please state your full name.

A. Frank C. Cherny.

Q. By whom are you employed?

A. I am employed by the U.S. Nuclear Regulatory Commission. I am a Section Leader in the Mechanical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation.

Q. Describe the nature of your work with respect to the Diablo Canyon operating license proceeding.

A. Task Coordinator for NUREG-0737 Item II.d.1 "Performance Testing of Boiling Water Reactor and Pressurized Water Ractor Relief and Safety Valve. The Diablo Canyon SER input for qualification of safety and relief valves, and block valves, as required by TMI Item I.D.1 was prepared under my supervision.

Q. Would you detail your professional qualifications?

A. Attached is a copy of my professional qualifications.

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Q. Have you, in the course of your professional experience, had occasion to review the design and qualification of relief, safety or block valves at Nuclear Power Plants other than Diablo Canyon?

A. Yes, I have reviewed such designs or assisted in their review. I have also participated in many meetings, discussions, design reviews of a generic nature, for the last five years, related to the design of overpressure protection systems for nuclear power plants. Many such activities were related to direct participation in industry standards writing activities associated with overpressure protection of nuclear reactor plant components and inservice testing of nucelar power plant pressure relief devices.

Q. In the course of your work on Diablo Canyon did you review the design and qualification of Reactor Coolant system relief valves?

A. Yes.

Q. In the course of your work on Diablo Canyon did you review the design and qualification of Reactor Coolant system safety valves?

A. Yes.

Q. In the course of your work on Diablo Canyon did you review the design and qualification of Reactor Coolant system block valves?

A. Yes.

Q. Are there General Design Criteria (GDC) which must be met which apply to relief and safety valves?

A. Yes, GDC 1, 14, 15 and 30 require the Applicant to assess their RCPB, including safety and relief valves, to meet certain standards.

Q. What standards of review does the Staff use when reviewing for compliance with GDC 1, 14, 15 and 30?

A. In reviewing for compliance with GDC 1, 14, 15 and 30 the following standards are used: (a) Standard Review Plan (SRP) 3.9.2, "Dynamic Testing and Analyses of Systems, Components, and Equipment."; (b) SRP 3.9.3, "ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures."; (c) Regulatory Guide 1.48 "Design limits and loading combinations for seismic Category 1 fluid systems components."; and (d) Regulatory Guide 1.68 "Pre-operational and Initial Startup Test Programs for Water Cooled Power Reactors."

Q. Would you briefly describe what "Standard Review Plan (SRP) 3.9.2., "Dynamic Testing and Analyses of Systems, Components, and Equipment" requires as regards reactor coolant system safety and relief valves?

A. Standard Review Plan 3.9.2 requires a (1) piping vibration preop. test program (2) seismic qualification of safety related mechanical equipment (3) dynamic system analysis to insure structural adequacy of piping loops for LOCA & SSE.

Q. Would you briefly describe what SRP 3.9.3 "ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures" requires as regards reactor coolant system safety and relief valves?

A. Standard Review Plan 3.9.3 (1) specifies what load combinations and stress limits apply (2) operability assurance program for "active" pumps and valves (3) design of pressure relief valve supports and associated discharge piping supports.

Q. Would you briefly describe what Regulatory Guide 1.48 "Design limits and loading combinations for seismic Category 1 fluid systems components" identifies as a standard as regards reactor coolant system safety and relief valves?

A. Regulatory Guide 1.48 delineates acceptable design limits and load combinations associated with normal operation and accident conditions.

Q. Would you briefly describe what Regulatory Guide 1.68 "Pre-operational and Initial Startup Test Programs for Water Cooled Power Reactors" identifies as a standard as regards reactor coolant system safety and relief valves?

A. Regulatory Guide 1.68 requires pre-operational testing of safety and relief valves to demonstrate that they will operate as required.

Q. What remains to be done by the Applicant to comply with GDC 1, 14, 15 and 30?

A. The tests performed to date do not cover loadings which result from transition flow from steam to water or solid fluid flow.

Q. How will the loadings which result from transition flow or solid fluid flow be addressed with respect to reactor coolant system safety and relief valves?

A. A test program has been initiated by the Electric Power Research Institute (EPRI) which will address safety and relief valve operability including loadings resulting from transition flow from steam to water and solid fluid flow. PG&E has committed to participating in this program and has as one of its objectives to satisfy the long-term requirements on SRV testing as set forth in Section 2.1.2 of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" included as of October 31, 1978 as Item II.D. in NUREG-0737. The Applicant has referenced the ongoing EPRI/NSAC safety and relief valve testing program. A description of the EPRI program was provided to NRC by EPRI in 1979 and an updated revision in July 1980. As noted in Supplement 14, the Staff is generally in agreement that the NUREG-0737 technical requirements for safety and relief valves can be met subject to receipt of additional information which was requested by letter of November 26, 1980 to Russell C. Youngdahl. By letter of December 15, 1980 EPRI responded to both the Staff's November 26, 1980 letter and NUREG-0737. PG&E has referenced the EPRI December 15 response in their February 26, 1981 letter.

The Staff has not completed its review of the December 15, 1980 letter primarily as regards EPRI proposed documentation submittal dates for safety and relief valves and comments made in the letter regarding resolution of a block valve test program after

July 1, 1981, the scheduled completion date for safety and relief valve testing.

Based on the Staff review of the EPRI program and PG&E's assurance that the EPRI program is applicable to the Diablo Canyon safety and relief valve designs, I believe there is adequate assurance that the NUREG-0737 requirement regarding performance verification of the RCS relief and safety valves will be met satisfactorily for Diablo Canyon.

Should this program demonstrate that these valves are not qualified for the above-stated loadings the Staff will require the licensee to take corrective actions. Present schedules indicate that this testing will be completed by July 1, 1981.

With regard to the safety valves, there is presently no evidence that these valves will not operate properly during the anticipated transients which produce transition flow and solid fluid flow.

Q. Other than with respect to transition flow and solid fluid flow, has the Applicant demonstrated compliance with the standards you identified as necessary to meet GDC 1, 14, 15 and 30?

A. Yes, compliance with Standard Review Plan (SPR) 3.9.2, "Dynamic Testing and Analyses of Systems, Components, and Equipment," at Diable Canyon which includes relief and safety valves is demonstrated in the Safety Evaluation Report on Diablo Canyon (SER) Section 3.9.1. and SER Supplements 7, 8, and 9.

Compliance with SRP 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures" at Diablo Canyon, which includes relief and safety valves, is demonstrated in SER Sections 3.9.2 and 5.2.1. (SER and SER Supplement 14).

Compliance with Regulatory Guide 1.48 "Design limits and loading combinations for seismic Category 1 fluid systems components" at Diablo Canyon, which includes relief and safety valves, is demonstrated in SER Section 5.2.1. (SER and SER Supplement 7 & 14).

Compliance with Regulatory Guide 1.68 "Pre-operational and Initial Startup Test Programs for Water Cooled Power Reactors" at Diablo Canyon, includes testing of relief and safety valves, is demonstrated in SER Section 14. (SER).

Compliance with the appropriate sections of Appendix B to 10 C.F.R. Part 50 at Diablo Canyon, including relief and safety valves, is demonstrated in SER Section 17.4. (SER).

Q. Is there anything, in your opinion, other than compliance with GDC 1, 14, 15 and 30, which demonstrates the reliability of the Reactor coolant system relief and safety valves at Diablo Canyon?

A. Yes. In addition to complying with the requirements of GDC 1, 14, 15 and 30 as discussed above. The reactor coolant system safety valves were originally designed and tested for operation on saturated steam in accordance with the applicable edition and addenda of Section III of ASME Boiler and Pressure Vessel Code.

Verification of this testing appears in the FSAR for Diablo Canyon, Table 5.2-1. As required by Article 9 of the ASME Boiler and Pressure Vessel Code, the safety valve relieving capacity has been provided so that the pressure limitation specified in the Code will be maintained under all of the system transients and accidents postulated to occur.

Q. Why do you believe that this further demonstrates the qualification of the reactor coolant system and safety valves?

A. Section III of the ASME Code provides specific valve functional requirements and installation requirements for the reactor coolant system safety valves. Additionally, it places restrictions on the types of pressure relief valves that can be used for such application and provides testing requirements for certifying the relieving capacity of the safety valves.

Q. Is there anything else you believe demonstrates the qualification of the reactor coolant system and safety valves?

A. Both safety valves and two of the three relief valves have been seismically qualified to be functional after exposure to loads resulting from the maximum hypothetical earthquake for Diablo Canyon as documented in Amendment 50 to FSAR Table 7-7. Also Safety and Relief valves will be operationally tested during the pre-op test program performed in accordance with Regulatory Guide 1.68.

Q. Why do you believe that these facts further demonstrate the qualification of the reactor coolant system relief and safety valves?

A. The pre-op testing of safety and relief valves demonstrates the operational readiness of the valve to lift within the prescribed set pressure range.

Seismic qualification assures that the valves will function should they be subjected to the maximum hypothetical earthquake which was postulated for Diablo Canyon.

Q. Are you familiar with I and E Bulletin 81-2?

A. Yes.

Q. Would you briefly summarize Bulletin 81-2?

A. I and E Bulletin 81-2 is entitled "Failure of Gate Valves to Close Against Differential Pressure". The Bulletin discusses valve closure tests recently performed by EPRI at the Marshall Test Facility on seven gate valves of the type commonly used as PWR PORV Block Valves. The testing included closing the valve against full flow steam differential pressure conditions selected as being representative of those that a PORV might be expected to close against. The Bulletin discusses the fact that three of the seven tested gate valves failed to fully close when subjected to the test conditions. It notes that valves of the type that failed are also supplied for utilization in a number of safety related applications in addition to the PORV block valve application where closure with a differential pressure across the valve is a requirement.

It requests that Licensees and Construction Permit Holders determine whether any of the "failed" valves are installed in such applications or planned to be installed. If no valves in this category are found this is to be reported to NRC. If one or more valves of this type are identified in such an application or intended for such an application, it imposes requirements for specific actions to be taken by both Licensees and Construction Permit Holders. Action to be taken include an evaluation of the significance of the valve failure to close on system operability in accordance with the plant technical specifications, modification of valves so they are qualified for the intended service or obtaining of qualified replacements.

Q. What type of valves are present at Diablo Canyon?

A. The Diablo Canyon plant has three Crosby HB-BP-86 (6M6) safety valves, three Masoneilan 20,000 Series (2 NPS) Power Operated Relief Valves and three Velan #B10-3054B013M Motor Operated Block Valves.

Q. Have any of these valves undergone the EPRI testing program?

A. As of May 5, 1981 the following testing had been performed by EPRI on valves of the type installed at Diablo Canyon:

Power Operated Relief Valve - A Masoneilan 20,000 Series PORV - full flow steam test.

#B10-3054B013M Velan Motor Operated Block Valve - full flow steam test. This valve is the same model as that used at Diablo Canyon.

Q. Were any of the block valves which failed the EPRI tests discussed in I and E Bulletin 81-2 of the type to be used at Diablo Canyon?

A. No.

Q. What testing remains to be completed on the valves at Diablo Canyon?

A. Additional testing of the Masoneilan Power Operated Relief Valve to include additional fluid effects that the valves could be exposed to under design basis transient or accident events such as the effects of loop seals and subcooled and saturated liquid is scheduled to be completed by July 1, 1981.

Complete testing of the Crosby HB-BP-86 (6M6) Safety Valve is scheduled to be performed during June of 1981 with a scheduled completion date of July 1, 1981.

The need for additional qualification testing of the Velan PORV Block Valve, as of May 5, 1981, is under discussion between PWR utilities and the NRC Staff. If additional testing is needed it must be completed by July 1, 1982 as specified in NUREG-0737.

Q. Will the remaining testing be completed prior to fuel load?

A. The qualification testing of Safety Valves and PORV's of the type installed on Diablo Canyon is scheduled to be completed by July 1, 1981, which will be well prior to fuel load.

Q. **Have any Diablo Canyon valves failed during the testing program?**

A. As of May 5, 1981, no valves of the type used in Diablo Canyon have failed any of the EPRI tests.

Q. What is done if a valve fails a test acceptance criterion?

A. EPRI has established a procedure so that all utilities participating in the program, the NSSS vendors, the valve manufacturers, and the NRC are all notified within a few days of any instances where a valve fails a test acceptance criterion. The NSSS vendors, with assistance from EPRI, assist the individual utilities with plant specific evaluations of the safety significance of any such failures. Depending on the results of these evaluations, actions are taken by the utilities in accordance with the regulations as regards reporting to NRC, possibly declaring equipment inoperable, if installed on an operating plant, and modifications or replacements of affected components for both operating plants and plants like Diablo Canyon that have a Construction Permit. The regulations also require the NSSS vendors and valve manufacturers to report safety related equipment anomalies. Additionally, the Office of Nuclear Reactor Regulation at the NRC independently reviews the details of all reported failures on a case by case basis and a decision is made as to what appropriate action should be taken.

For safety and relief valves of the type installed at Diablo Canyon, as noted above, testing is scheduled to be completed by July 1, 1981. If any failures of valves of this type occur in the EPRI testing program, NRC will require the effects of the specific failure on safe operation of the Diablo Canyon to be expeditiously evaluated by PG&E on a schedule such that any necessary modifications or replacements of safety or relief valves can be made prior to initial fuel loading.

As noted above, it is not clear at this time whether additional testing is required to confirm the capability of the type of PORV Block Valve installed at Diablo Canyon to open and close against all fluid conditions that could result from design basis transients and accidents. If more testing is required, as specified in NUREG-0737, it must be completed by July 1, 1982.

If it is determined that additional testing of block valves is required to confirm their performance capability, a procedure will be established for expeditious handling of adverse test results. Valve modifications or replacements, if any are warranted, will be made to the Diablo Canyon PORV Block Valves on a schedule consistent with the safety significance of any observed anomalies.

Q. In view of the above testimony, do you have an opinion as to whether fuel loading and low power testing can commence at Diablo Canyon while PORV Block Valves remain to be tested?

A. As noted in this testimony, it has not been determined that additional testing will be required to confirm the opening and closing capability of the Diablo Canyon type block valves.

Based on the fact that the Diablo Canyon safety and relief valves will be fully qualified prior to fuel loading for service conditions far in excess of those conditions valves could be exposed to during low power testing, the other factors discussed aboe, and on the testimony of Norman Lauben, it is my opinion that fuel loading and low power testing can commence at Diablo Canyon with no adverse affect on the health and safety of the public.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PACIFIC GAS AND ELECTRIC COMPANY	)	Docket Nos. 50-275 O.L.
	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant Unit Nos. 1 and 2)	)	

FRANK C. CHERNY

PROFESSIONAL QUALIFICATIONS

MECHANICAL ENGINEERING BRANCH

DIVISION OF ENGINEERING

I am a Mechanical Engineer in the Mechanical Engineering Branch responsible for the review and evaluation of design criteria of mechanical components, of methods of dynamic analysis and testing of safety related systems and components and of criteria for protection against dynamic effects associated with postulated failures of fluid system components for nuclear service.

I graduated from Marquette University with a B.S. degree in Mechanical Engineering in 1965.

From July 1965 to November 1968 I was employed by the Babcock & Wilcox Co. at offices in both Barberton and Akron, Ohio. During the majority of this period I was engaged in materials engineering work, primarily writing technical ordering requirements for primary pressure boundary materials to be used for reactor vessels, steam generators, and presurrizers for both commercial and U. S. Navy nuclear systems. In addition I had assignments of several months duration each in quality control engineering and nuclear steam supply system performance engineering.

From November 1968 to May 1974 I was employed in the Pressurized Water Reactors Division of Westinghouse Nuclear Energy Systems. My work experience during this period includes the following:

From November 1968 to May 1970 and September 1970 to April 1971 as a Reactor Vessel Project Engineer based in Monroeville, Pa.:

(1) I had overall project engineer responsibility for design and construction of reactor vessels for several Westinghouse nuclear power plants in the U.S. My responsibilities included preparation of Design Specifications and review of vendor documentation for compliance with Westinghouse, Utility, ASME, Architect Engineer and AEC requirements. I was personally responsible for coordination of the technical aspects of the transfer of two partially completed reactor vessels from a U.S. manufacturer's shop to a European manufacturer for completion when schedular problems developed at the U.S. manufacturer.

(2) After the U.S.-Europe transfer of these components, I assumed responsibility for technical coordination between the primary Westinghouse nuclear engineering office in the U.S. and an overseas office established in Brussels, Belgium to do project engineering work for mechanical components used in Westinghouse nuclear plants both in the U.S. and in Europe.

From June 1970 to September 1970 and from April 1971 to December 1972 I was employed by Westinghouse Nuclear Energy Systems in Europe based in Brussels, Belgium. My responsibilities included:

(1) During the June-September, 1970 period and from April 1971 to about April 1972 I had project engineer responsibility for several reactor pressure vessels and a pressurizer. I also acted as Westinghouse engineering representative for U.S. AEC Quality Assurance audits of European vessel manufacturers.

(2) From April 1972 to December 1972 I served as a lead engineer with a broader scope of responsibility. I was responsible for reactor pressure vessels, pressurizers and reactor vessel supports fabricated in Europe for Westinghouse Nuclear Plants. Several engineers and a technician reported directly to me during this period. The work included preparation of Design Specifications, review and approval of vendor design and manufacturing documentation, and coordination with both U.S. and European utility and regulatory representatives.

From December 1972 to May 1974 I was again based in Monroeville, Pa., this time as Senior Reactor Vessel Project Engineer. I was responsible for the technical adequacy of several reactor pressure vessels being manufactured in the U.S. for use in Westinghouse Nuclear Plants in Europe. I was also responsible, during the majority of this period, for the training of a Westinghouse Nuclear - Europe engineer temporarily based in the U.S.

In May of 1974 I started work for the Regulatory Division of the U.S. Atomic Energy Commission and have remained through the transition into the U.S. Nuclear Regulatory Commission. In 1977 I was appointed as a Section Leader in the Mechanical Engineering Branch. As a branch technical reviewer and later as a Section Leader I have been participating in the review of construction permit and Operating License applications.

Since July of 1974 I have served as a member of the ASME Section III Subgroup On Pressure Relief which is responsible for writing industry standards for the overpressure protection of light water reactor plant components. Additionally, since 1977 I have been a member of the ASME Working Group on Safety and Relief Valves. The Working Group has recently completed work on a proposed industry standard entitled "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices." The proposed standard was issued for public comment by ASME early in 1981.

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(Diablo Canyon Nuclear Power Plant	)	
Unit Nos. 1 and 2)	)	

TESTIMONY OF G. NORMAN LAUBEN

Q. Please state your full name.

A. G. Norman Lauben.

Q. By whom are you employed, and describe the work you perform?

A. I am employed by the Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. A copy of my professional qualifications is attached to this testimony.

Q. Have you read "Memorandum and Order (Granting PG&E's and NRC Staff Motions for Summary Disposition of Joint Intervenor's Contentions 5 and 13; Denying their Motions as to Contentions 4 and 24)"? In particular, have you read Contentions 4 and 24 as stated therein?

A. Yes.

Q. Would you describe the scope of the subject matter addressed in your testimony?

A. I have been asked to address the safety significance of the low power testing program as it would affect the necessity to have in

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place an emergency plan which meets all the requirements of NUREG-0654. I have also been asked to address the safety significance of block valve failure during the low power testing program.

Q. What are the major factors affecting safe reactor operation during low power testing?

A. In supplement 10 to the Diablo Canyon SER (NUREG-0675), the risk associated with low power testing was addressed. As discussed in that SER there are three major factors which contribute to a substantial reduction in risk for low power testing as compared to continuous full power operation. First, there is additional time available for the operators to correct the loss of important safety systems needed to mitigate relatively high risk events, or to take alternate courses of action. Secondly, supplement 10 also discussed the significant events that contribute to that risk and the reduction in risk associated with the significant postulated events during the low power testing program. Third, there is a reduction in required capacity for mitigating systems at low power.

Q. What are the significant postulated events that could potentially affect the public health and safety?

A. The dominant events are (1) small break LOCAs with loss of the emergency core cooling system (ECCS), (2) transients with total loss of feedwater, and (3) failure of double check valves between the reactor coolant system (high pressure) and the residual heat removal system (low pressure) which results in a LOCA (inter-system LOCA) outside containment, i.e. the interior of the reactor vessel communicates directly with the environment. Since the publication of the Reactor Safety Study (WASH-1400), the NRC staff has continued to study the risk to the public from potential severe accidents at nuclear power plants. This effort has confirmed that the event scenarios dominating accident risks are generally the same for different PWR designs, i.e., small break loss of coolant accidents (LOCAs) and transients.

Q. What was done specifically to address the question of risk due to low power testing, and what was the conclusion of that effort?

A. We have reexamined the dominant scenarios to estimate the reduction in the probability of the event because of the additional time available during low power operation for the reactor operators to correct the loss of important safety systems needed to mitigate the event or to take alternate courses of action. Similarly, we have calculated the reduced fission product inventory for operation of an initially unirradiated core at 5% power for 6 months and have determined the reduction in potential public exposure via reduction in potential release magnitudes. Risk is roughly proportional to the probability of severe accidents (which lose the heat sink) and to the fission product inventory in the core. From these factors we have estimated that the overall reduction in risk to the public should be a factor of 400 to 1500 if a plant is operated at 5% power from initial startup for 6 months compared to continuous full power operation.

Q. In your review of fuel load and low power test programs what have you concluded about the potential risk of low power testing?

A. Based on the actual power history of other reactors during their low power testing program, the actual power history expected at Diablo Canyon would result in even less available fission product inventory. The peak power during this time period is only expected to be 3 or 4% of rated capacity. Operation at this power level is only expected for a maximum of 10 days. This would result in a further risk reduction by a factor of about 2. It is therefore concluded that the public risk due to fuel loading and the proposed low power test program is less than public risk due to full power long-term operation by a factor of about 400 to 3000.

Q. How could risk to the public be affected by small break or inter-system LOCA's?

A. Risk to the public would occur only if there is release of substantial amounts of radioactive fission products outside the containment. This could occur only if there is a failure to cool the core for an extended period of time. During this time the fuel element cladding would have to fail by overheating. The reactor coolant pressure boundary would have to be violated and the reactor building containment would have to be violated. Of course the latter condition would occur for the intersystem LOCA. The important factor is that the core would have to remain uncooled for a significant length of time. (Approximately 10 hours during the low power test program).

However, NRC requires that all facilities licensed to operate are provided with reliable and redundant emergency core cooling systems (ECCS). NRC regulations (10 C.F.R §50.46) require applicants to analyze a spectrum of pipe breaks and locations with various assumed equipment failures. These analyses are performed with NRC-specified conservative assumptions and must demonstrate coolability of the core and minimum generation of hydrogen. Specifically, ECCS evaluations must demonstrate that ECCS performance will result in a coolable geometry and less than 1% core-wide metal-water reaction even with the reactor at 102% power and worst-case linear heat rates. The Diablo Canyon ECCS is required to conform to these requirements. Thus, for all power levels the requirements of NRC regulations provide adequate protection against severe core damage. Thus for small break LOCAs, substantial risk to the health and safety of the public would occur only if the ECCS failed to operate as designed. Therefore, when performing risk assessment this condition must be evaluated.

Q. With a reactor operating at a maximum of 5% of full power, could these events lead to the significant amounts of core damage?

A. No. I have looked at these events and have concluded, as discussed herein, that at 5% power it is extremely unlikely that such events would lead to significant amounts of core damage by "significant" I mean 5% metal-water reaction.

Q. Please provide the basis for your conclusions?

A. LOCA analyses with severly degraded ECCS's were performed which demonstrate the large amount of time that would be available at 5% power for diagnosis and corrective action to prevent significant core damage. For these analyses, it was conservatively assumed that none of the pumped ECCS systems was functioning. I have had Sandia Laboratories, our consultant, run a calculation for me using the RELAP4 code to estimate the time at which boil-off begins. For a small 4-inch cold-leg break LOCA, boil-off would not begin for about 1 hour and uncovery would be delayed until about 3 hours. Rapid increase in fuel clad temperature, leading to severe core damage, would not begin until about 15 hours.

I have also obtained information from the NRC staff Project Manager for the Sequoyah nuclear power plant concerning the actual maximum power level and test duration for the low-power test program. Sequoyah is a comparable facility and the test program is similar to that planned for Diablo Canyon. This information indicates that actual test power was about 4% full power or less and lasted only some 8 days. For similar conditions at Diablo Canyon uncovery would not occur until about 4 to 5 hours. More than 20 hours would elapse before significant core damage would occur.

Q. Based upon this assessment, what is the likelihood of significant core damage at low power due to a LOCA?

A. As I have indicated above, with the NRC's requirements for reliable ECCS performance, the potential for severe core damage and associated significant hydrogen generation is very small even at full power. The time available at low power for the operator to take corrective action ranges from 15 hours to more than 20 hours in the event of a small LOCA. In addition, the coolant flow required to dissipate decay heat at 10 hours following a LOCA would be only about 8 gpm which is within the capacity of the centrifugal charging pump used for the normal make-up systems. Because of the time available for the operators to correct malfunctions in the ECCS or to initiate cooling with the normal charging system, we believe that the probability of a small LOCA resulting in excessive fuel damage and significant radiological release is reduced by at least a factor of 400 to 1600 for low power operation as compared to operation at full power.

Q. What about the potential effects of other significant events, e.g., transients with total loss of feedwater, on the public risk?

A. At 5% power the number of events that can result in failure to adequately cool the core is greatly reduced. All transients initiated by turbine trip are eliminated since the turbine is not on line. Total loss of feedwater caused by any other transient becomes negligible with respect to core damage and public risk. In such a case, core heat is transferred through the steam generators from the primary to secondary systems. After a scram from 5% power I calculate that it would take about 2½ days to boil the steam generators dry, conservatively assuming no feedwater makeup. During that period of time, diagnosis, corrective action or alternate heat removal methods could easily be accomplished. Moreover, by that time fission product heat will have decayed sufficiently so that passive steam heat losses (radiant heat transfer) would be enough to keep the reactor cool, even if no corrective action were taken. As a consequence we believe that the total risk reduction associated with these events is about 1000 to 20,000 lower compared to full power operation.

Q. If, as you say, feedwater transients are of almost no concern at low power, what about other scenarios? Could they not now become dominant at low power?

A. Other transients (steam line break, steam generator tube rupture, rod ejection and ATWS) were also examined. A similar reduction in risk was evident. Therefore, these transients did not become dominant.

Q. But an ATWS event could result in a significant primary to secondary heat imbalance. Why isn't this of concern?

A. If one assumes that the highly unlikely scenario of total loss of feedwater is followed by a failure of the reactor system to scram (i.e., the worst ATWS event), complete boil-off of the water in the steam generators would occur in 45 minutes. During this period of time there are a number of things the operator could do to bring the reactor to safe shutdown, including initiation of the boron injection system and diagnosis and correction of the failure to scram. These would terminate the event before boil-off of significant reactor vessel-inventory and thus, well before the onset of severe core damage. Moreover, at low power, significant overpressurization of the primary system does not occur because of the low integrated reactor power. It should be emphasized that the sequence which involves a transient, total loss of feedwater, and failure to scram as independent events has such a low probability that it cannot be considered a credible event (less than 10^{-7} per reactor year).

Q. How does the above discussion relate to the need for a qualified emergency plan during low power operation?

A. The above discussion shows that abundant time (at least 20 hours) is available to take corrective action to mitigate or terminate the most likely scenarios which could affect public risk during low power testing. For some sequences of concern at full power, no action would be required during low power operation to prevent public risk. Under these conditions the risk is so small that there is virtually no need for a qualified emergency plan.

Q. What is the safety significant of a failure of a block valve during low power testing?

A. None. Each block valve is located upstream of one PORV. There are three sets of these valve pairs in parallel. In theory a block valve could fail open or closed. During operation the block valves are normally open and the PORVs are closed. If the PORVs are not challenged the position of the block valves has no significance. PORVs can be challenged only if a heat imbalance occurs. As discussed earlier the transient events that could cause this condition are those for which secondary heat sink is lost when feedwater is lost. At low power these events have virtually no safety significance and PORV challenge is extremely unlikely.

Q. But what if a block valve and a PORV should both fail open?

A. The likelihood of this double failure is extremely remote. As noted in the testimony of Mr. Frank Cherny, the Diablo Canyon type PORVs will be tested in the EPRI program prior to fuel loading. If this should occur it would simply be a small LOCA which was discussed previously. The size of this LOCA would be even smaller than that previously discussed. Thus even more time would be available to correct, mitigate or avert fuel damage and public risk.

STATEMENT OF PROFESSIONAL QUALIFICATIONS

NORMAN LAUBEN

My name is George Norman Lauben. I am employed as a Nuclear Engineer in the Reactor Systems Branch, Division of Systems Integration, U.S. Nuclear Regulatory Commission. I have worked in the field of nuclear reactor for 19 years, and in nuclear activities for 23 years. I have worked for the Commission and its predecessor, the Atomic Energy Commission, since 1968. During this time I have worked directly on reactor safety matters, including Emergency Core Cooling System (ECCS) performance review and Loss-of-Coolant Accident (LOCA) analysis.

I was a member of the 1971 AEC ECCS task force and the AEC Staff Panel for the ECCS Rulemaking Hearing. I am the author of the TOODEE2 computer program used by the NRC and the nuclear industry for transient fuel pin thermal analysis during a LOCA. I was a member of the technical team that accompanied Mr. Harold Denton to the Three Mile Island Reactor on March 30, 1979.

I have a B.S. and M.S. in Chemical Engineering from Case Institute of Technology (now Case Western Reserve University).

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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	)	50-323 O.L.
(Diablo Canyon Nuclear Power Plant	)	
Unit Nos. 1 and 2	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of TESTIMONY OF BARTHOLOMEW C. BUCKLEY, JOHN SEARS, FRANK C. CHERNY, AND NORMAN LAUBER in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, this 8th day of May, 1981.

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