UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL 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, Units 1 and 2)	)	

TESTIMONY ON BEHALF OF THE INDEPENDENT
DESIGN VERIFICATION PROGRAM

OF

Mr. John E. Krechting
Dr. William E. Cooper

REGARDING

CONTENTION 4.a.-l. and o.-u.

In the Matter of:

PACIFIC GAS AND ELECTRIC COMPANY

(Diablo Canyon Nuclear Power Plant, Units 1 and 2)

Docket Nos. 50-275 O.L.
50-323 O.L.

TESTIMONY REGARDING CONTENTIONS 4.a.-l. and 4.o.-u.

INTRODUCTORY TESTIMONY

Q.1: Please state your name, current position, business address and qualifications.

A.1: This information is contained in A.1. of the Testimony Regarding Contentions 1,2 and 5-8.

Q.2: Please describe your participation in the Independent Design Verification Program (IDVP).

A.2: This information is contained in A.2 of the Testimony Regarding Contentions 1,2 and 5-8.

Q.3: What is the purpose of your testimony?

A.3: Contention 4 alleges that the IDVP "accepted deviations from the licensing criteria without providing adequate engineering justifications" in a number of specific respects. This testimony addresses every subpart of Contention 4, except Contentions 4.m. and 4.n., which are addressed in the Testimony of the Panel addressing Contentions 3,4.m. and 4.n.

Q.4: Does every answer in this testimony constitute the testimony of both members of the panel?

A.4: Yes. Since Mr. Krechting had the responsibility for the technical review by the IDVP of each of the subject areas covered by the testimony, he is more familiar with the details of such review. However. Dr. Cooper had overall responsibility for program management of the IDVP, reviewed and approved the disposition of the EOIs which are referred to in the testimony, and shares the judgments expressed in the testimony.

CONTENTION 4.a.

"Contrary to the requirements of FSAR Section 17.1 regarding compliance of the as-built installation with the design documents, the IDVP review of the AFWS disclosed that the as-built installation failed to meet the design drawings in that (i) a steam trap on the turbine-driven AFW pump steam supply line is not provided and (ii) there are discrepancies in the arrangement of the long-term cooling water supply line."

Q.1: Does the IDVP know the origin of allegation (i) in this contention?

A.1: The IDVP believes that the origin of allegation (i) is EOI 8027.

Q.2: Please describe the issue there identified.

A.2: The IDVP's concern was that a steam trap shown on a piping schematic drawing it was reviewing had not been installed.

Q.3: How was this concern resolved?

A.3. After issuance of EOI 8027, the IDVP determined that although the design originally did not call for the steam trap, a design change had been subsequently initiated adding the steam trap which appears in a piping schematic drawing reviewed by the IDVP. However, the design change document was not signed by General Construction (G.C.) to authorize installation because it was subsequently determined by start-up testing that the trap was not required. Therefore, the design change adding the steam trap was never officially approved. G.C. wrote a design change document superseding the original and revising the piping schematic.

Thus, the IDVP determined that the as-built condition (without the steam trap) corresponds to the approved design. The IDVP review is reported in ITR-22.

Q.4: Does the IDVP know the origin of allegation (ii)?

A.4: The IDVP believes that the origin of allegation (ii) is EOI 8048.

Q.5: Please describe the issue there identified.

A.5: The IDVP's concern was that a check valve not shown on a piping schematic drawing it was reviewing had been installed in the long-term cooling water supply line.

Q.6: How was this concern resolved?

A.6: After EOI 8048 was issued, the IDVP determined that the long-term cooling water supply line had a check valve, as the original design required. A design change to that supply line had been issued which did <u>not</u> require removal of the check valve. However, a draftsman misinterpreted a Xerox copy of the design change document and incorrectly removed the check valve from the piping schematic drawing. This error on the drawing was corrected.

Thus, the IDVP verified that the as-built installation corresponds to the approved design. The IDVP review is reported in ITR-22.

Q.7: Did the IDVP accept any deviations from licensing criteria relating to the as-built installation of the AFWS?

A.7: No. No deviation from the licensing criteria exists.

CONTENTION 4.b.

"Contrary to FSAR Section 8.3.3, the electrical design does not fully comply with the commitments regarding separation and color coding."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origins are EOIs 8055 and 8059.

Q.2: Please describe the issue identified in EOI 8055.

A.2: In EOI 8055, the IDVP's concern was that two pressure indicators (P153A and P153B) on the main control board did not meet the separation criteria of FSAR, Section 8.3.3, in that the indicators are less than five inches apart.

Q.3: How was this concern resolved?

A.3: After issuance of EOI 8055, PGandE stated that the intent of the Section 8.3.3 separation criteria is to provide adequate isolation and insulation between exposed current-carrying portions of mutually redundant power control devices, e.g., transfer switches. However, these separation criteria were not intended to apply to low energy instrumentation signal devices such as the pressure indicators which were addressed in EOI 8055. The IDVP accepted PGandE's interpretation of the FSAR, Section 8.3.3, as reasonable and consistent with the underlying basis for this section, subject to PGandE's commitment to revise Section 8.3.3 to clarify the requirements for separation of mutually redundant indicating devices on the main control board. The IDVP review is reported in ITR-27.

Q.4: Please describe the issue identified in EOI 8059.

A.4: The IDVP interpreted FSAR Section 8.3.3 to provide for the color-coding of safety-related cables only. Since the IDVP found that some non-safety-related cable was color-coded, it issued EOI 8059 to assure that the matter would be clarified.

Q.5: How was this concern resolved?

A.5: After issuance of EOI 8059, PGandE explained that the intent of FSAR Section 8.3.3 was to ensure that safety-related cable was identified but that it did not preclude color-coding of some non-safety-related cable. The IDVP accepted PGandE's interpretation of the FSAR, Section 8.3.3 as reasonable and consistent with the underlying basis for this section, subject to PGandE's commitment to revise Section 8.3.3 to clarify the color-coding requirements. The IDVP review is reported in ITRs-27 and -28.

Q.6: Did the IDVP accept any deviations from licensing criteria relating to separation of electrical circuits and devices or color-coding?

A.6: No. There were no deviations from licensing requirements.

CONTENTION 4.c.

"Contrary to the single failure criterion of Appendix A to 10 CFR Part 50, a single failure may cause loss of redundant power divisions because redundant electric power division trains are electrically interconnected through two circuit breakers and a single power transfer switch."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origin is EOI 8041.

Q.2: Please describe the issue identified in EOI 8041.

A.2. The IDVP was concerned with the possibility of an improper transfer from normal to alternate sources of electric power through use of a switch common to both sources at a location in the CRVP system. Failure of the switch could cause damage to two non-mutually redundant safety-related circuits.

Q.3: How was that concern resolved?

A.3. After issuance of EOI 8041, PGandE demonstrated that its standard operating practice for transfer switches allows connection of only one of the two sources at any time. PGandE issued a formal operating order for DCNPP-1 which specifies its standard transfer procedure. Therefore, the highly improbable failure of the switch will not affect both non-mutually redundant circuits.

Q.4: Did the IDVP accept any deviations from the single failure criterion with respect to a power transfer switch?

A.4. No. There is no deviation from licensing requirements. Operator action ensures proper switch/breaker operation. The IDVP review is reported in ITR-26.

CONTENTION 4.d.

"Contrary to GDC 57 of Appendix A, valve operators for the isolation valves which provide the steam supply to the turbine-driven auxiliary feed pump from two of the main steam generators have not been classified and procured as safety-related components."

Q.1: Can the IDVP address this contention?

A.1: No.

Q.2: Why not?

A.2: Review of the valves in question with respect to GDC 57 was not within the scope of the IDVP sample as described in the Phase II Program Plan.

CONTENTION 4.e.

"The single failure of an auxiliary relay would prevent automatic closure of the redundant steam generator blowdown isolation valves on automatic initiation of the AFWS contrary to a Westinghouse interface requirement and FSAR Figure 7.2-1."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origins are EOI 8047 and SER Supplement 18, p. C.4-12.

Q.2: Please describe the issue identified in EOI 8047.

A.2. The IDVP was concerned that failure of a specific non-safety grade relay (3 AFWP) would prevent the automatic closure of the steam generator blowdown valves. The IDVP questioned whether continued blowdown from the four steam generators during a postulated accident requiring auxiliary feedwater system operation had been considered in the accident analysis in FSAR Chapter 15.

Q.3: How was this concern resolved?

A.3: The IDVP subsequently determined that various analyses by Westinghouse of accidents requiring operation of the AFW system had assumed that the blowdown valves are isolated. Results of these analyses were forwarded to the NRC (letter of October 9, 1980, Crane of PGandE to Schwencer of NRC). The assumption of blowdown isolation is used in the analysis which supports the conclusions of FSAR Sections 15.2.8 and 15.2.9. However, PGandE satisfactorily demonstrated that, for the accidents described in FSAR Sections 15.2.8 and 15.2.9, if protection systems do not initiate a diverse signal to trip safety-grade blowdown valves, adequate auxiliary feedwater flow exists assuming both a single failure of one AFW train and blowdown valves unisolated. In other Chapter 15 accident scenarios requiring auxiliary feedwater, the IDVP verified that the blowdown valves would be tripped closed by safety-grade trip signals from diverse sources (such as safety injection). In these cases, the blowdown valves would receive diverse safety-grade trip signals and close in accordance with the Westinghouse accident analysis assumptions.

Q.4: Did the IDVP accept deviations from applicable licensing criteria relating to the failure of an auxiliary relay?

A.4: No. As stated in ITR-27, the IDVP concluded that no safety limits or licensing commitments have been violated with regard to the ability to mitigate accidents or remove decay heat and cool down the plant described in Chapter 15 of the FSAR. FSAR Figure 7.2-1 is a functional drawing which depicts the signals that close the blowdown valves; in the IDVP's opinion it does not reflect a Westinghouse interface requirement for redundant relays. This interpretation has been confirmed by a letter from Westinghouse to PGandE dated September 6, 1983. In the opinion of the IDVP, the Westinghouse interface requirements are satisfied.

CONTENTION 4.f.

"Contrary to NUREG 0588 regarding environmental qualifications, flow transmitter FT-78 and flow control valve FCV-95 are located in a harsh environnment but were not listed as such in the PG&E Environmental Qualification Report dated September 1981, and are not yet environmentally qualified."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origin is EOI 8052.

Q.2: Please describe the issue identified in EOI 8052.

A.2: The IDVP expressed concern that the specified transmitter and valve were not qualified for a harsh environment.

Q.3: How was this concern resolved?

A.3: PGandE responded that the identification tag for FT-78 was changed to FT-200. FT-200 is listed in the Environmental Qualification (EQ) Report. PGandE also indicated that this flow transmitter is qualified for a harsh environment based on the vendor's report which justifies operation pending completion of the qualification program. PGandE stated that the item is in the vendor's on-going qualification program and that qualification documentation will be added when complete.

In addition, PGandE responded that the EQ Report fails to identify FCV-95 as being in a harsh environment but that, in fact, it has been qualified for a harsh environment. PGandE provided its component evaluation report to document its approval of the vendor's qualification testing.

The NRC's DCNPP-1 SER, Supplement No. 15, states that it performed a 100% review of PGandE's equipment qualification program and found that it meets regulatory requirements. The Supplement also records that NRC acknowledges and accepts the fact that equipment can be conditionally qualified or that additional information may be needed to complete qualification.

Q.4: Did the IDVP accept deviations from the licensing criteria relating to the environmental qualification of this transmitter and valve?

A.4: No. On the basis of the IDVP's review (reported in ITR-27) and the NRC SER assessment, the IDVP concludes that there are no deviations from licensing criteria.

CONTENTION 4.g.

"Contrary to the requirements of NUREG 0588 regarding environmental qualifications, portions of the CRVPs were omitted from PG&E's Environmental Qualification report."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origin is EOI 8056.

Q.2: Please describe the issue identified in EOI 8056.

A.2: The IDVP was concerned that some CRVP system Class IE equiment had not been identified in the EQ Report.

Q.3: How was this concern resolved?

A.3: Further verification determined that some Class IE equipment for the CRVP system was not listed in the EQ Report because that list was compiled prior to preparation of the schematic drawings showing the modified system. However, the equipment in the CRVP system which was not listed in the EQ Report will operate in and was designed for a mild environment. The IDVP identified no CRVP Class IE equipment not listed in the EQ Report which is required to be qualified for a harsh environment. The IDVP review is reported in ITR-26.

Q.4: Did the IDVP accept deviations from applicable licensing criteria relating to environmental qualification of portions of the CRVP?

A.4: No. The IDVP concluded that the CRVP equipment meets the environmental qualification requirements of NUREG-0588 and no deviation from licensing criteria exists.

CONTENTION 4.h.

"Contrary to PG&E's September 14 and December 28, 1978 licensing commitments, CRVPS equipment identified in the FSAR as necessary to maintain control room habitability during safe shutdown has not been evaluated regarding the effects of a moderate energy pipe break."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origin is EOI 8050.

Q.2: Please describe the issue identified in EOI 8050.

A.2: The PGandE letters to the NRC referenced in Contention 4.h. described the moderate energy line break (MELB) evaluation performed for DCNPP-1. In EOI 8050, the IDVP recorded its concern that those letters did not identify the CRVP system among those being evaluated.

Q.3: How was this concern resolved?

A.3: In response to EOI 8050, PGandE provided the IDVP with an evaluation of the effects of MELB on the CRVP system. This evaluation indicated that a MELB could cause loss of one CRVP system train. An assumed single failure of the redundant CRVP system train could then degrade control room habitability. However, if the control room should become uninhabitable, the capability for plant shutdown and cooldown would be available from the hot shutdown panel.

The IDVP verification confirmed that, in the unlikely event that a MELB caused the control room to become uninhabitable, plant shutdown and cooldown capability could be maintained from the hot shutdown panel. PGandE stated that the CRVP system was not included in the MELB evaluation because of its conclusion that there was no need to evaluate the CRVP system since, even if the system failed, the plant could be shut down from the hot shutdown panel. The IDVP judged that the PGandE conclusion was reasonable and acceptable. The IDVP review is reported in ITR-21.

Q.4: Did the IDVP accept a deviation from applicable licensing criteria relating to evaluation of CRVP equipment regarding the effects of an MELB?

A.4: No. The IDVP concluded that there is no deviation since plant safe shutdown capability is not impaired as a result of MELBs which could affect the CRVP system.

CONTENTION 4.1.

"The fire protection for the motor driven AFW pump room is not consistent with the PG&E licensing commitment for fire zone separation as stated in its November 13, 1978 Supplemental Information for Fire Protection Review ("SIFPR") in that:

1. There is a large grated ventilation opening in the ceiling of the room;
2. a fire damper has gaps when it is closed."

Q.1: Does the IDVP know the origin of these allegations?

A.1: The IDVP believes that the origins are EOIs 8038 and 8037.

Q.2: Please describe the issue identified in EOI 8038.

A.2: The IDVP's concern was whether a ventilation opening was clearly identified in the FSAR definition of the barrier between the fire zones here involved (FSAR Amendment 51, p. 4-18).

Q.3: How was the concern resolved?

A.3: EOI 8038 was issued because the FSAR language was subject to misinterpretation if taken literally. However, review of postulated credible fires indicated that a fire in one zone will not propagate through the opening to the other zone. Thus, the SIFPR licensing commitment that a fire will not propagate from one fire zone to another is satisfied and the requirement (FSAR, Amendment 51, p. 5-4) that plant safe shutdown is not hindered is met. The IDVP review is reported in ITR-18.

Q.4: Please describe the issue identified in EOI 8037.

A.4: In EOI 8037, the IDVP's initial concern was whether fire damper FD-24 was UL qualified, and that it had air gaps.

Q.5: How was this concern resolved?

A.5: Subsequent IDVP verification determined that the fire damper is UL qualified and that the damper gaps satisfy vendor design and UL qualification requirements. The IDVP review is reported in ITR-18.

Q.6: Did the IDVP accept deviations from fire protection licensing criteria?

A.6: No. The IDVP concluded that no deviations from licensing criteria exist.

CONTENTION 4.j.

"The fire protection for the AFW pump room is not consistent with the PG&E licensing commitment for cable separation as stated in its SIFPR of November 13, 1978 in that:

1. the pumps for the motor driven AFW pumps and the control circuitry for a flow control valve necessary for operation of the turbine driven AFW pump are located in a single fire zone;
2. cables for some AFW circuits are not routed in accord with descriptions in the SIFPR and four AFW circuits PG&E committed to identify and review in the SIFPR were not included in that document."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origins are EOIs 8019 and 8021.

Q.2: Please describe the issue identified in EOI 8019.

A.2: IDVP was concerned that circuits for the motor-driven AFW pumps and the control circuitry for a flow control valve (FCV-95) necessary for operation of the turbine-driven AFW pump were located in a single fire zone.

Q.3: How was this concern resolved?

A.3: Further verification determined that control circuitry for FCV-95 was not located in this fire zone. Therefore, no violation of separation requirements occurred and a single fire cannot prevent proper operation of the AFW system. The IDVP review is reported in ITR-18.

Q.4: Please describe the issue identified in EOI 8021.

A.4: The IDVP's concern was that as-built circuit routings were different than indicated in the SIFPR.

Q.5: Was that correct?

A.5: Yes. Cables had been re-routed subsequent to issuance of the SIFPR.

Q.6: Did the re-routing violate the requirement for cable separation?

A.6: No. The IDVP subsequently field verified that all AFW system as-built circuit routing conformed with the licensing commitment for separation, with the exception that the FCV-95 DC circuit was improperly located at that time. However, the FCV-95 circuit was then in the process of being re-routed for reasons other than fire protection considerations and installation was not complete. That installation was subsequently completed and the IDVP verified that the completed routing of the FCV-95 circuit also conforms with licensing separation commitments. The IDVP review is reported in ITR-18.

Q.7: Did the IDVP accept a deviation from cable separation requirements for the AFW system?

A.7: No. There is no deviation from separation requirements and a single fire could not prevent proper operation of the AFW system.

CONTENTION 4.k.

"Contrary to the licensing commitment set forth in its SIFPR of November 13, 1978 each of the three 4160 volt cable spreading rooms has a ventilation opening leading up to the 4160 volt switchgear rooms."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origin is EOI 8039.

Q.2: Please describe the issue identified in EOI 8039.

A.2: The IDVP's concern was whether a ventilation opening was identified in the FSAR definition of the barrier between the fire zones here involved (FSAR Amendment 51, p. 4-45).

Q.3: How was this concern resolved?

A.3: EOI 8039 was issued because the FSAR language was subject to misinterpretation if taken literally. Further review of postulated credible fires indicated that a fire in any of these rooms would be unlikely to propagate through the opening to the other fire zone and that, even if a fire did propagate through the opening, it would affect only one vital bus and safe shutdown capability would not be affected. The IDVP review is reported in ITR-18.

Q.4: Did the IDVP accept deviations from fire protection licensing criteria?

A.4: No. The IDVP concluded that a single fire in any of the 4160 V cable spreading or switchgear rooms would not adversely affect plant safe shutdown capability. No deviation from licensing criteria exists.

CONTENTION 4.1.

"Contrary to FSAR Section 3.6, possible jet impingement loads have not been considered in the design and qualification of safety-related piping and equipment inside containment."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origins are EOI 7002 and SER Supplement 18, C.4-29.

Q.2: Please describe the issue identified in EOI 7002.

A.2: The IDVP's concern was that no objective evidence could be found in the PGandE files that the effects of jet impingement on components inside containment were considered.

Q.3: How was this concern resolved?

A.3: The DCP performed and documented a reanalysis of the jet impingement effects of HELB inside containment. The IDVP verified the DCP efforts on a sampling basis to ensure that the DCP sufficiently documented its jet impingement reanalysis such that the concern of EOI 7002 could be resolved. The IDVP verification included review of the DCP reanalysis procedure; review of the DCP field review, including an independent walkdown to verify DCP identification of jet impingement interaction with safety-related targets; and review of DCP safety evaluation for jet impinged targets. The IDVP verification is reported in ITR-48.

The IDVP sample verified that assumed failure of instrumentation, instrument tubing, electrical components and electrical conduits identified as jet impingement targets did not negate the ability to mitigate the effects of the specific jet impingment from the causing HELB. For safety-related structural and mechanical jet impinged targets, including pipes, piping and equipment supports, structural beams and columns, concrete walls and floors, the IDVP verified that the DCP had applied the applicable threshold loads, if any, in accordance with the DCP jet impingement reanalysis procedure and criteria or, as required, that the DCP had identified these targets for further analyses. These analyses would determine the structural adequacy of these targets to withstand the loads from the postulated HELB jet impingement. The IDVP determined that it did not need to review these further analyses to resolve EOI 7002, because EOI 7002 did not identify detailed structural and pipe load analysis as a concern.

In the SER, Supplement 18, the NRC identified impingement loads on safety-related piping and equipment inside containment as an open item. PGandE has responded to this open item in a letter to the NRC, dated September 9, 1983. This letter summarizes the specific DCNPP-1 licensing commitments of the FSAR and PGandE's compliance with those commitments.

Q.4: Did the IDVP accept any deviations from licensing criteria with respect to jet impingement resulting from HELBs inside containment?

A.4: No. The IDVP concluded that the DCP reanalysis procedure and criteria met the licensing commitments of FSAR, Section 3.6, constituted a comprehensive technical review program, and documented the technical approach and the results.

CONTENTION 4.o.

"Contrary to the requirements of NUREG-0588 regarding environmental qualifications, safety-related cables and cable splices which could be subject to a harsh environment during a high-energy line break are not identified in the PG&E Environmental Qualification Report."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origins are EOIs 8011 and 8044.

Q.2: Please describe the issue identified in EOI 8011.

A.2: The IDVP's concern was that some safety-related cables in the AFW and CRVP systems were not identified as environmentally qualified in Reference 5 of Appendix 3.6 of the FSAR.

Q.3: How was this concern resolved?

A.3: PGandE subsequently provided documentation showing that all safety-related cables used in the AFW and CRVP systems are environmentally qualified. It should be noted that reference 5 had been prepared in 1975. The IDVP verified that cable not listed in that document was purchased after 1975, qualified to the temperature defined in Appendix 3.6 of the FSAR, and included in the plant EQ Report. The IDVP review is reported in ITRs-21, -25 and -26.

Q.4: Please describe the issue identified in EOI 8044.

A.4: The IDVP's concern was that, although Reference 5 of FSAR Appendix 3.6 states that "in general splices were not used," the IDVP found splices which could potentially be exposed to the effects of HELCs.

Q.5: How was this concern resolved?

A.5: In response to EOI 8044, PGandE provided documentation showing that the splices in question are qualified to 340^{O}F. The

IDVP verified that no splice in the AFW or CRVP system is exposed to jet temperature in excess of 340^{o}F and that splices are addressed in the EQ Report. The IDVP review is reported in ITRs-25 and -26.

Q.6: Did the IDVP accept deviations from licensing criteria with regard to environmental qualification of cables and cable splices?

A.6: No. The IDVP concluded that all cables and splices in the AFW and CRVP systems are environmentally qualified in accordance with NUREG-0588.

CONTENTION 4.p.

"The NSC pipe break analysis, which is Appendix A to FSAR Section 3.6, did not include all likely sources of water in the calculation of flooding levels."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origins are EOIs 8005 and 8040.

Q.2: Please describe the issue identified in those EOIs.

A.2: The IDVP's concern was that the water levels calculated to occur in area GE/GW at elevation 115 feet, 0 inches due to a feedwater HELB were not conservative.

Q.3: How was this concern resolved?

A.3: Further detailed review of the PGandE calculation revealed the maximum flood heights which it calculated for the area in question are conservative because PGandE used conservative assumptions and methods. Although the volume of water below the feedwater sparger ring (inlet pipe) in the steam generator and from the AFW system were neglected in PGandE's calculation, the IDVP determined that other sources of water were overpredicted by substantially greater amounts. This resulted in conservative predictions by PGandE of water release volumes and flood heights. The IDVP review is reported in ITR-14.

Q.4: Did the IDVP accept deviations from the licensing criteria for flooding?

A.4: No. The licensing criteria of FSAR 3.6, Appendix A are met.

CONTENTION 4.q.

"Contrary to PG&E's December 28, 1979 licensing commitment letter to the NRC, modifications to protect two Auxiliary Feedwater valves from the effects of moderate energy line breaks were not implemented."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origin is EOI 8014.

Q.2. Please describe the issue identified in EOI 8014.

A.2. The IDVP's concern was that two valves (FCV-436 and FCV-437) identified in the December 28, 1979 letter as requiring protection from the effects of MELBs did not in fact have spray shields installed.

Q.3: How was this concern resolved?

A.3: PGandE demonstrated that, subsequent to the December 28, 1979 letter, it had determined that these two valves are not required to operate in mitigation of the effects of a MELB. PGandE therefore decided not to install the spray protection devices. PGandE will revise the December 28, 1979 letter to indicate that protection for these valves is not required. The IDVP performed an evaluation and determined that the valves are in fact not required to operate to mitigate the effect of an MELB. The IDVP review is reported in ITR-21.

Q.4: Did the IDVP accept deviations from applicable licensing criteria with respect to protection from MELBs?

A.4: No. There is no deviation from applicable licensing criteria.

CONTENTION 4.r.

"Contrary to the licensing commitment to maintain minimum system redundancy as stated in FSAR Section 3.6A (NSC evaluation of pipe break outside containment), four components were identified for which high energy line cracks could cause temperatures in excess of the specification temperatures of the components."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origins are EOIs 8028, 8029, 8030 and 8031.

Q.2: Please describe the issues identified in EOIs 8028, 8029 and 8030.

A.2: The IDVP identified inconsistencies in Appendix 3.6 of the FSAR and Reference 5 of this Appendix. The Appendix states (pp. 3.6A-22 and -23 (Revision 3)) that HELBs (or HELCs) need not be postulated in line 760 downstream of FCV-95. The Appendix (at pp. 3.6A-68 and 3.6A-82) and Reference 5 (Table B-13) indicate that breaks were postulated in line 760 downstream of FCV-95.

Q.3: How was this issue resolved?

A.3: The applicable NRC requirements are found in a letter from Mr. Giambusso of the NRC to Mr. Searls of PGandE dated December 18, 1972. The IDVP verified that the letter does not require postulation of HELBs or HELCs in line 760 downstream of FCV-95. PGandE has therefore, committed to correct the inconsistencies in the FSAR by eliminating any such postulation. Therefore the equipment identified in these EOIs will not be exposed to harsh environments. The IDVP review is reported in ITR-21.

Q.4: Please describe the issue identified in EOI 8031.

A.4: The IDVP's concern was that an HELC in line 594 could adversely affect AFW system equipment.

Q.5: How has this issue been resolved?

A.5: Further verification indicated that a crack in line 594 would not cause turbine generator or reactor trip. Thus, in accordance with the FSAR criteria one need not assume loss of offsite power and the AFW system is not required to operate to mitigate the effects of the HELC in line 594 or to achieve plant shutdown. Therefore, the AFW equipment identified in this EOI will not be required to operate to mitigate the effects of the HELC in line 594 or to safely shut down the plant. Accordingly, there is no safety consequence if the AFW equipment is exposed to HELC impingement temperatures. The IDVP review is reported in ITR-21.

Q.6: Did the IDVP accept deviations from applicable licensing criteria regarding HELCs?

A.6: No. There is no deviation from FSAR Appendix 3.6. Minimum system redundancy is maintained and equipment required to mitigate the HELC and safely shut down the plant will not be exposed to temperatures in excess of its specification temperature.

CONTENTION 4.s

"Contrary to the licensing commitment to maintain minimum system redundancy as stated in FSAR, Section 3.6A (NSC evaluation of pipe break outside containment), a conduit was identified whose failure due to a high energy line crack could eliminate redundant Auxiliary Feedwater system flow."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origin is EOI 8049.

Q.2: Please describe the issue identified in EOI 8049.

A.2: The IDVP's concern was that the effects of an HELB jet on the AFW system conduit had not been considered.

Q.3: How was this issue resolved?

A.3: The IDVP verified that the cable in the conduit identified in the EOI will not be damaged due to the effects of an HELB and that AFW system redundancy is not affected. Jet pressures on the conduit are lower than allowable conduit jet pressure. Cable in the conduit is qualified to $540^{0}F$, which is above the enveloping temperature using either the FSAR methodology or ANSI-ANS 58.2 methodology. The IDVP review is reported in ITR-23.

In addition, since a break in line 594 will not cause a turbine-generator or reactor trip, the FSAR does not require an assumption of loss of offsite power and AFW system operation is not required to mitigate effects of the break or to shut the plant down. Therefore, the AFW system is not required to mitigate the HELB in question or to safely shut down the plant.

Q.4: Did the IDVP accept deviations from licensing criteria in evaluating HELB effects on AFW system conduits?

A.4: No. There is no deviation from licensing criteria.

CONTENTION 4.t.

"Contrary to the FSAR Section 8.3 commitment to provide switchgear buses with adequate short circuit interrupting capability, the calculated duties for circuit breakers on 4160 V buses F, G, and H were above the nameplate ratings for those buses."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origin is EOI 8022.

Q.2: Please describe the issue identified in EOI 8022.

A.2: The IDVP's concern was that the circuit breaker's nameplate current-interrupting rating was less than the calculated current-interrupting duty required.

Q.3: How was this concern resolved?

A.3: In response to EOI 8022, PGandE provided the manufacturer's verification that the 4160 V circuit breakers are capable of interrupting the maximum available short circuit current. This conclusion is based on tests performed by the manufacturer in 1976. The IDVP review is reported in ITR-24.

Q.4: Did the IDVP accept a deviation from licensing criteria in connection with short circuit current-interrupting capability?

A.5: No. The IDVP concluded that, since the breakers will interrupt the calculated short circuit current, no deviation from licensing criteria exists.

CONTENTION 4.u.

"Contrary to single failure criteria stated in FSAR Section 3.1.1, reviews of the Auxiliary Feedwater and Control Room Ventilation and Pressurization systems identified circuit separation and single failure deficiencies. Similar deficiencies were identified in additional verification reviews, which included other safety-related systems."

Q.1: Does the IDVP know the origin of this contention?

A.1: The IDVP believes that the origins are EOIs 8017 and 8057.

Q.2: Please describe the issues identified in EOIs 8017 and 8057.

A.2: The IDVP review of the initial sample systems (AFW and CRVP systems) identified Class IE electrical control circuits in enclosures (i.e., panels and termination boxes) that were not separated by the methods listed in the FSAR, Section 8.3.3. This issue is the subject of EOI 8017. The IDVP also identified an electrical control transfer switch in the CRVP system to which mutually redundant Class IE power sources were connected such that the DCNPP-1 single failure criterion was not satisfied. This issue is the subject of EOI 8057.

Q.3: Were these issues generic?

A.3: The IDVP believed the issues were generic.

Q.4: How were these concerns resolved?

A.4: Since these concerns were considered by the IDVP to be generic, the DCP reviewed all PGandE-designed safety-related systems for similar concerns. The DCP review identified all mutually redundant Class IE circuits and devices within the same enclosure that required separation. The DCP then conducted a field review of all those identified circuits to ensure that they are separated by the methods stated in the FSAR, Section 8.3.3. Also, as part of the review, the DCP identified electrical devices which have mutually redundant circuits connected to them. If such an electrical device was identifed, a single failure analysis was performed by the DCP to establish the ability of the system to perform its design basis function.

Upon completion of the DCP analysis, the IDVP selected four of the PGandE-designed safety-related systems as samples to verify the DCP review. The IDVP reviewed each sample system's circuit drawings to determine whether the DCP had correctly identified the mutually redundant circuits. The IDVP then performed a field inspection of enclosures included in the sample systems to determine whether the installation of mutually redundant circuits within the same enclosure met the separation criteria committed to in FSAR Section 8.3.3. The IDVP verification did not identify any cases where the DCP had failed to identify a system's mutually redundant circuits or where separation in the field did not meet FSAR requirements. The DCP had identified modifications that were required to be made in the sample systems. The IDVP field verified that those modifications had been implemented such that compliance with the FSAR separation criteria exists.

In addition to verifying circuit separation, the IDVP also reviewed the DCP's single failure analyses in the four sample systems. The DCP provided drawings marked to show where mutually redundant circuits were connected to the same device. In those cases, the DCP performed a single failure analysis to establish the ability of the system to perform its design basis function. The IDVP reviewed these drawings and analyses to verify that all mutually redundant circuits connected to the same device had been identified. The IDVP verification did not identify any case where a single failure could adversely affect the operation of a sample system and no modificatons were required. The IDVP review is reported in ITRs-27, -28 and -49.

Q.5: Did the IDVP accept deviations from licensing criteria with respect to circuit separation and single failure criterion as applied to safety-related systems?

A.5: No. The concerns identified in EOIs 8017 and 8057 have been eliminated in all PGandE-designed safety-related systems. Circuit separation and single failure requirements of FSAR, Sections 8.3.3 and 3.1.1, are met.

AFFIDAVIT OF WILLIAM E. COOPER

The undersigned, William E. Cooper, this 12th day of October, 1983, upon his oath states that the attached Resume is a true and correct statement of his education and professional experience.

William E. Cooper

October 12, 1983

Notary Public

WILLIAM S. MOONAN
NOTARY PUBLIC
MY COMMISSION EXPIRES
AUGUST 6, 1987

TELEDYNE ENGINEERING SERVICES

ATTACHMENT 1

DR. WILLIAM E. COOPER
Consulting Engineer

Resume

Education

Stevens Institute of Technology (1941-1943)
Oregon State College, U.S. Army, Spec. Training, M.E. 4-7 (1943-1944)
Oregon State College (1946-1948):
B.S. in Mechanical Engineering (1947)
M.S. in Mechanical Engineering (1948)
Purdue University (1948-1951):
Ph.D. in Engineering Mechanics (1951)

Honors

Fellow, American Society of Mechanical Engineers (1972)
Purdue Distinguished Engineering Alumnus (1973)
Certificate of Appreciation, Pressure Vessel Research Committee (1977)
The William M. Murray Lectureship,
Society for Experimental Stress Analysis (1977)
B. F. Langer Nuclear Codes and Standards Award,
American Society of Mechanical Engineers (1978)
Centennial Award, American Society of Mechanical Engineers (1980)
Pressure Vessel and Piping Medal,
American Society of Mechanical Engineers (1983)
Sigma Xi (Research), Pi Tau Sigma (Mechanical Engineering),
Sigma Pi Sigma (Physics)
Who's Who in: America; Engineering; Atoms
American Men and Women of Science

Registered Professional Engineer

Indiana (1952), New York (1958), Massachusetts (1963)

Membership

American Society of Mechanical Engineers
Society for Experimental Stress Analysis
Atomic Industrial Forum

Addresses

Business: Teledyne Engineering Services
130 Second Avenue
Waltham, Massachusetts 02254
(617) 890-3350

Home: 83 Fifer Lane
Lexington, Massachusetts 02173
(617) 861-7007

Employment

Teledyne Engineering Services
(formerly Teledyne Materials Research or Lessells and Associates, Inc.):
Consulting Engineer (1976-)
Senior Vice President & Technical Director (1974-1976)
Vice President & Manager, Engineering (1968-1973)
Engineering Manager (1963-1968)
Consulting engineering services in the design and analysis of mechanical systems and structures, primarily for energy conversion.

Massachusetts Institute of Technology, Lecturer, Reactor Safety (1975-)

Electric Power Research Institute, Consultant (1974-1978)

Oak Ridge National Laboratory
Advisory Committee, Engineering Technology Division (1975-1978)

Advisory Committee on Reactor Safeguards, Consultant (1967-1974)

Knolls Atomic Power Laboratory, General Electric Company:
Consulting Engineer, S5G Structural Mechanics (1963)
Manager, S5G Structural Evaluation (1959-1963)
Consulting Engineer, SAR Structural Evaluation (1957-1959)
Specialist, SAR Mechanical Analysis (1955-1977)
Engineer, SIR Mechanical Analysis (1952-1954)
Technically responsible for the structural integrity of components and systems of sodium- and water-cooled naval reactor power plants.

Union College
Instructed graduate courses in Theory of Plasticity (1962 and 1963)

Purdue University:
Instructor in Engineering Mechanics (1949-1952)
Instructor in Engineering Drawing (1948-1949)
Instructed courses in drafting, statics and dynamics, experimental stress analysis, plasticity, dynamics of materials, physical metallurgy, and applied metallography

Oregon State College:
Graduate Teaching Assistant in Engineering Drawing (1947-1948)
Student Teaching Assistant in Physics (1947)

U.S. Army, Sergeant, Construction Foreman (1943-1946)

General Electric Company, Mechanical Draftsman (1942-1943)

Committee Participation

American National Standards Institute:

Board of Directors (1981-)
Nuclear Standards Management Board (1975-1977)
Technical Advisory Group to TC/85, Nuclear (1974-1979)
International Standards Organization
U.S. Representative to TC/85, SC/3, Nuclear Power (1976-1979)
Expert to TC/85, SC/3, WG/6 Primary Boundary (1974-)

American Society of Mechanical Engineers:

Senior Vice President and Chairman, Council on Codes and Standards (1981-)
Vice President for Codes and Standards and Member Executive Committee of Council (1980-1981)
Committee on Budget (1978-1980) (Chairman 1979-1980)
Council (formerly Policy Board) on Codes and Standards (1972-)
(Chairman 1980-)
Nuclear Codes and Standards Committee (1974-1980) (Chairman 1975-1977)
Boiler and Pressure Vessel Committee:
Honorary Member (19[illegible]-)
Main Committee (196[illegible]-1980)
Executive Committee (1971-1976)
SC on Design (1967-1975) (Chairman 1967-1972)
SC on Nuclear Certification (1973-1977) (Chairman 1973-1975)
SC on Nuclear Power (1964-1980) (Vice Chairman 1966-1969)
Special Committee to Review Code Stress Basis (1955-1967)
B31 Code for Pressure Piping:
Mechanical Design Committee (1957-1959, 1963-1967)
B31.7 Nuclear Piping (1965-1971)
Code for Nuclear Pumps and Valves (1965-1969)
Metals Engineering Division, Chairman (1960)
Hudson-Mohawk Section, Chairman (1958-1959)

Atomic Industrial Forum:

Subcommittee on Materials Requirements (1981-)

Welding Research Council:

Pressure Vessel Research Committee:
Main Committee (1954-1974)
Design Division (1954-1974) (Chairman 1969-1973)

Publications and Major Presentations

"Determination of Principal Plastic Strains," Transactions, ASME, July 1952.

"Structural Problems of a Sodium-Cooled Nuclear Reactor," ASME, Paper No. 54-SA-25, with D. R. Miller.

"Proposed Structural Design Basis for Nuclear Reactor Pressure Vessels, Problems in Nuclear Engineering," edited by D. J. Hughes, S. McLain, C. Williams, Pergamon Press, 1954.

"The Significance of the Tensile Test to Pressure Vessel Design," Welding Journal Research Supplement, January 1957.

"Experimental Determination of Stresses in the Vicinity of Pipe Appendages to a Cylindrical Shell," Proceedings SESA, XIV, 2, with F. J. Mehringer, 1957.

"Safeguards Aspects of Reactor Vessel Design," The Welding Journal Research Supplement, January 1958; Journal American Society of Naval Engineers, with D. R. Miller, May 1958.

"The Scope of Pressure Vessel Codes and Activities Towards Improved Content," Preprint 78, Nuclear Engineering and Science Congress, 1958.

"Design Basis for Thermal Stress," Proc. SESA, XV, 2, 1958.

"Structural Design Basis for Reactor Pressure Vessels and Associated Components," U.S. Office of Technical Services PB151987, with B. F. Langer (Westinghouse) and J. L. Mershon (BuShips), December 1958.

"Implications of Radiation Effects to Reactor Pressure Vessel Design, AEC Conference on the Status of Radiation Effects Research on Structural Materials and the Implications to Reactor Design," October 1959.

"Stresses in a Pipe Bent into a Circular Arc," Transactions, ASME, Journal of Engineering for Industry, 83, B, 4, with N. A. Weil and J. E. Brock, November 1961, pp. 449-459.

"Specification Guidelines for Nuclear Pressure Vessels," USAEC NYO-3416-1, with D. F. Landers, October 1964.

"Design Criteria for High-Pressure, High-Temperature Bolting," Nuclear Engineering and Design, 8, with R. Widmer, J. A. Signorelli, R. F. Brodrick, 1968, p. 125.

Publications and Major Presentations (Cont'd)

"Interaction of Material and Design Problems in Critical Vessels," Invited Keynote Lecture, First International Conference on Pressure Vessel Technology, Delft, Proceedings, Part III, 1969, p. 29.

"Construction, Rating, and Inservice Inspection of Test Tanks," Proceedings, 7th U.S. Navy Symposium of Military Oceanography, Vol. 1, with B. H. Schofield, 1970, p. 104.

"Experimental Efforts on Bursting of Constrained Disks as Related to the Effective Utilization of Yield Strength," ASME Paper 71-PVP-79, with E. H. Kottcamp and G. A. Spiering.

"Codes: Asset or Liability," Fatigue at Elevated Temperature, ASTM STP 520, 1972.

"Development and Operation of the ASME Boiler and Pressure Vessel Code," and "An Introduction to the Design Procedures of the ASME Boiler and Pressure Vessel Code," U.S.-Japan Joint Symposium, Pressure Vessel Technology and Pressure Component Codes, Tokyo, 1973.

"Nuclear-Pressure Vessels and Piping-Materials: Where to Next?" ASME Joint Conference, Miami, 1974.

"A Personal Viewpoint on the Development of ASME Code Rules for Nuclear Components," ASME Winter Annual Meeting, 1974.

"Nuclear Vessels are Safe," Mechanical Engineering, with B. F. Langer, April 1975.

"Improving Reactor Pressure Vessel Availability by Design," Nuclear Safety, 17(1), January-February 1976.

"ASME Section XI Flaw Evaluation Procedures and Application to Nozzles," Nondestructive Examination Conference, Washington, 1976, also UKAEA, Risley, 1976.

"Experimental Mechanics and Nuclear Power," The William R. Murray Lecture, 1977, Experimental Mechanics, 17, 10, October 1977.

"Safety Evaluation of Reactor Vessel Nozzle Cracks," ASME Paper No. 78-PVP-90, with P. C. Riccardella, 1978.

"Minimization of Safety and Reliability Concerns by Consideration of Operating Experience," Conference on the Quality of Nuclear Power Stations from American and German Viewpoints, Köln, 1978.

Publications and Major Presentations (Cont'd)

"Analysis of Inservice Inspection Flaw Indications," Maintenance Welding in Nuclear Power Plants, American Welding Society, 1979.

"The Future - an ASME Viewpoint," ANS Executive Conference on Nuclear Power Plant Owner Certification, Washington, 1980.

"Concepts in the Design and Analysis of Welded Joints," AWS, Indianapolis, 1980.

"What Happened to Common Sense," ASME Emerging Technologies Conference, San Francisco, 1980.

"The Development of Codes and Standards for Superconducting Magnet Structures," DOE-NBS Workshop on Materials at Low Temperatures, Vail, 1980.

"International Involvement of U.S. Standards," U.S. Department of Commerce Conference, 1980.

"Owner Certification," Atomic Industrial Forum Workshop on Reactor Construction and Operation in the New Environment, Atlanta, 1980.

"Requalification of Nuclear Class 1 Pressure Boundary Components, SMIRT, Paris, 1981 (also EPRI Report NP-1921).

AFFIDAVIT OF ROBERT L. CLOUD

The undersigned, Robert L. Cloud, this 12th day of October, 1983, upon his oath states that the attached Professional Resume is a true and correct statement of his education and professional experience.

Robert L. Cloud

Robert L. Cloud

October 12, 1983

OFFICIAL SEAL
PATRICIA L. HOLMES
NOTARY PUBLIC-CALIFORNIA
COUNTY OF ALAMEDA
My Commission Expires Sept. 6, 1985

Patricia L. Holmes

Notary Public

Robert L. Cloud Associates, Inc.

ROBERT L. CLOUD

PRINCIPAL

Professional Resume

Education

Texas A & M College	BSME	1956
Texas A & M College	MSME	1957
Univ. of Pittsburgh	Phd ME	1964

Experience

1979 to Present: Robert L. Cloud Associates, Inc., Berkeley, Ca. Design Criteria, Seismic design and analysis, Piping design criteria, Piping analysis, Project management, Failure analysis

1978-1979: Engineering Decision Analysis Co., Palo Alto, Ca., Exec. Vice President. Project management, Design criteria, Failure analysis, Piping and Mech. Equipment design and analysis.

1971-1978: Westinghouse Electric Corp., PWR Systems Division; Manager of Mechanics and Materials Technology. Responsible for design criteria, stress and dynamic analysis and materials engineering for the primary system of Westinghouse Pressurized Water Reactor Systems.

1969-1971: Teledyne Materials Research, Waltham, Mass. Manager Analytical Engineering, Design criteria, Analysis and research on equipment and piping, Failure analysis.

1962-1969: Westinghouse Electric Corp., Bettis Atomic Power Lab. Stress Analysis Engineer to Manager, Mechanics and Materials Engineering, Design criteria, Fracture Mechanics Studies, Analysis and research on pressure vessels and piping.

1957-1962: Westinghouse Electric Corp.. Large Rotating Apparatus Division. Stress analysis and development work on large central station turbo-generators.

1956-1957: Texas A & M University Instructor, Mechanical Engineering.

Membership

1. American Society of Mechanical Engineers
   a) Past Chairman, Design and Analysis Committee, PVP Division
   b) Past Chairman, Pressure Vessels and Piping Division
   c) Past Member, ASME Boiler and Pressure Vessel Code, Subgroup on Openings and Attachments

2. Past member, Pressure Vessel Research Committee, WRC

Lectures

1. Eisenment Lectures, Fracture Control, 1970, American Society for Metals, Philadelphia, Pa.

2. Teledyne Materials Research
   ASME Boiler and Pressure Vessel Code Seminar
   a) Brittle Fracture
   b) Nozzles, Tubesheets, & Special Problems
   c) Plastic Limit Analysis

3. Principal Division F. Lecture, "Structural Mechanics Applied to Pressurized Water Reactor Systems", 4th International Conference on Structural Mechanics in Reactor Technology, San Francisco, California, 1977.

Publications

"Minimum Weight Design of a Radial Nozzle in a Spherical Shell,: Transactions of the ASME, Journal of Applied Mechanics, Vol. 32, Series E. No. 2, June, 1965.

"The Limit Pressure of Radial Nozzles in Spherical Shells" Nuclear Structural Engineering, Vol. 1, No. 4, April 1965.

"Interpretive Report on Pressure Vessel Heads:, Welding Research Council, Bulletin No. 119, January 1967.

"Approximate Analysis of the Plastic Limit Pressure of Nozzles in Cylindrical Shells" with E.C. Rodabaugh, Transactions of the ASME, Journal of Engineering for Power, Vol. 90, Series A, No. 2, April 1968.

Thermal Buckling and Frictional Effects on Postbuckling Behavior of Sealed Electric Liners" with J.H. Dittmar, Transactions of the ASME, Journal of Engineering for Industry, Vol. 90, Series B, No. 3, August, 1968.

"Assessment of the Plastic Strength of Pressure Vessel Nozzles" with E. C. Rodabaugh, Transactions of the ASME Journal of Engineering for Industry, Vol. 90, Series B, N. 4, November, 1968.

"Evaluation of Experimental and Theoretical Data on Radial Nozzles in Pressure Vessels" with E. C. Rodabaugh R. J. Atterbury, and F. J. Witt, U.S. Atomic Energy Commission, TID - 24342, 1968.

"Proposed Reinforcement Design Procedure for Radial Nozzles in Cylindrical Shells with Internal Pressure" with E. C. Rodabaugh, Welding Research Council Bulletin No. 133, September 1968.

"Fracture Mechanics Criteria for the Prevention of Brittle Fracture in Nuclear Reactor Vessels," 1967, (Classified) with others, Bettis Atomic Power Lab., Westinghouse Electric Corporation.

"Pressure Vessel Head Design" chapter in "The Stress Analysis of Pressure Vessels and Pressure Vessel Components" Editor, S. S. Gill, Pergamon Press, 1970.

"Fracture Prevention in Nuclear Plants" ASM Conference on Fracture Control, Philadelphia, Pennsylvania, 1970.

Editor, "Pressure Vessels and Piping: Design and Analysis", 2 Vol., American Society of Mechanical Engineers, 1972.

"Dynamic Analysis of Nonlinear Pipe Whip Restraints" with S. Palusamy, and W. L. Patrick, Pressure Vessels and Piping Conference, Miami Beach, Florida, June 1974.

"Nonlinear Seismic Analysis of the Ice Condenser System" with W. S. LaPay, A. J. Soroka, and G. J. Bohm, Structural Design of Nuclear Plant, ASCE 1975 New Orleans, Louisiana.

"Dynamic Analysis of Structures with Solid-Fluid Interaction" with R. R. Pedrido, A. N. Nahavandi, Transactions of the 4th International Conference on Structural Mechanics in Reactor Technology (Smirt-4), San Francisco, California, August, 1977.

"Structural Mechanics Applied to Pressurized Water Reactor Systems", Vol. 46, No. 2, Nuclear Engrn. & Design, April, 1978.

"Dynamic Events in Nuclear Reactors", Survival of Mechanical Systems in Transient Environments, T. L. Geers et al, Editors, ASME AMD-Vol. 36, 1979.

"Creep Instability in Flexible Piping Joints" with R. D. Campbell and D. Bushnell, 1980. To be published.

"Seismic Performance of Piping in Past Earthquakes:, Specialty Conference on Civil Engineering and Nuclear Power, September 1980, Knoxville, Tenn.

"A Summary and Critical Evaluation of Stress Intensity Factor Solutions of Corner Cracks at the Edge of a Hole" with S. S. Palusamy, Welding Research Council Bulletin No. 276, April 1982.

"Interpretive Report on Dynamic Analysis of Pressure Components - Second Edition", Chapter 3, Welding Research Council Bulletin No. 269, August 1981.

AFFIDAVIT OF JOHN E. KRECHTING

The undersigned, John E. Krechting, this 12th day of October, 1983, upon his oath states that the attached Resume is a true and correct statement of his education and professional experience.

John E. Krechting

John E. Krechting

October 12, 1983

Elizabeth M Doll

Notary Public

ATTACHMENT 3

August 1983

KRECHTING, JOHN E.

PROJECT ENGINEER
POWER DIVISION

EDUCATION

U.S. Naval Academy - Bachelor of Science, Naval Science 1965

LICENSES AND REGISTRATIONS

Professional Engineer - Rhode Island

EXPERIENCE SUMMARY

Mr. Krechting has over 18 years of experience in the engineering field. Currently as Project Engineer for the Diablo Canyon Nuclear Power Plant Independent Design Verification Program, he is responsible for the NRC required design verification to establish that installed safety-related systems meet their licensing and operational commitments.

Since joining Stone & Webster Engineering Corporation (SWEC) in July 1974 as an Engineer in the Power Division, Mr. Krechting has been assigned to positions of increasing responsibility. He has been assigned to the Charlestown Nuclear Power Plant project, which was in the design development and PSAR production stage; to the high temperature gas-cooled reactor (HTGR) 3,000 MWt Reference II Design Study for General Atomic Company which developed a conceptual reference plant design; and to the Sundesert Nuclear Plant project which was in the design development and PSAR production stage. Mr. Krechting was assigned as the Principal Nuclear Engineer on the North Anna Power Station - Units 1 and 2 project, and subsequently as the Lead Power Engineer on the North Anna Power Station project. He was assigned as Supervisor of the Systems Engineering Group responsible for the development and maintenance of fluid system descriptions for the SWEC reference/standard nuclear, fossil, and industrial plants; development and maintenance of fluid system related Power Division Technical Procedures and Guidelines; and resolution of generic fluid system design problems.

Prior to joining SWEC, he was employed by Westinghouse Nuclear Energy Systems as a Senior Systems Engineer on the project to determine the feasibility of floating nuclear power plants. He developed the design of many of the nuclear and reactor auxiliary systems for the Offshore Power Systems' floating nuclear power plants.

His experience includes 6 years in the operation and maintenance of U.S. Navy submarine nuclear power plants, including two years as the Chief Engineering Officer of a nuclear submarine power plant.

DETAILED EXPERIENCE RECORD
KRECHTING, JOHN E. 50109

STONE & WEBSTER ENGINEERING CORPORATION, BOSTON, MA (July 1974 to Present)

Appointments:

Supervisor, Systems Engineering Group - July 1980
Senior Power Engineer - March 1979
Power Engineer - December 1977
Engineer, Power Division - July 1974

Diablo Canyon Nuclear Power Plant, Pacific Gas and Electric Company
(Nov 1982 to Present)

As PROJECT ENGINEER (Nov 1982 to Present), directly responsible for the safety-related system design portion of the NRC mandated Independent Design Verification Program (IDVP) for the Diablo Canyon Nuclear Power Plant (DCNPP). The project is unique because it is the first and most comprehensive IDVP required by the NRC. Responsibilities include the technical supervision of the mechanical, electrical and instrumentation and control verification of selected safety-related systems. Responsible for the analysis to develop environmental temperatures and pressures due to high energy line break outside the containment. Also responsible for staffing; establishing and meeting schedules, estimating and controlling costs; and maintaining client and NRC liaison.

As LEAD POWER ENGINEER (June 1982-Nov 1982), directly responsible for the Independent Design Verification of the mechanical and nuclear design of selected safety-related fluid and HVAC systems. Responsibilities included technical and administrative supervision of Power Division Engineers assigned to the project.

Systems Engineering Group, Power Division (July 1980-Nov 1982)

As SUPERVISOR of the Systems Engineering Group, directly responsible for development of Reference Fossil Power Plant (RFPP) fluid systems design, including preparation and maintenance of system descriptions and P&ID's; development of Reference Nuclear Power Plant (RNPP) fluid systems design, including preparation and maintenance of system descriptions and P&ID's; development of the Industrial Reference Power Plant (IRPP) fluid systems design, including preparation and maintenance of system descriptions and P&ID's; development and maintenance of system-related Power Division Technical Procedures and Guidelines; and resolution of nuclear and fossil plant fluid system related generic engineering and design problem reports and development of preferred solutions.

North Anna Power Station - Unit 2, Virginia Electric and Power Company
(Aug 1977-July 1980)

As LEAD POWER ENGINEER (June 1978-July 1980), directly responsible for the supervision and administrative control of all Power Division personnel assigned to the 900 MWe project, including nuclear, mechanical, facilities and piping Engineers and Designers; technical responsibility for the power plant's nuclear systems, steam plant systems, and HVAC systems, including equipment and piping arrangements, conformance to design codes, performance calculations, and drawings; preparation and technical adequacy of nuclear, steam plant and HVAC equipment and process specifications; coordination and approval of project work performed by the Power Division staff groups; development of engineering man-hour estimates and schedules to ensure timely completion of work; and coordination of interface between the Power Division and other engineering disciplines, such as Structural, Electrical, Engineering Mechanics, and Control Divisions.

As PRINCIPAL NUCLEAR ENGINEER (Aug 1977-June 1978), directly responsible for the technical design of the plant's nuclear and nuclear auxiliary systems, including piping arrangements, conformance to design codes, and preparation of design calculations. Also responsible for the supervision and coordination of the Engineers in the Nuclear Engineering Group, including scheduling of work and preparation of nuclear equipment specifications and purchase orders.

Sundesert Nuclear Power Plant, San Diego Gas & Electric Company
(Jan 1976-Aug 1977)

As ENGINEER on the 900 MWe project, directly responsible for coordination of the layout of the annulus building to ensure compliance with system design criteria, conformance with NRC high energy line criteria, optimization of space utilization, and development of layout requirements. Developed PSAR write-ups for the NSSS systems, including reactor coolant system, chemical and volume control, residual heat removal, and safety injection. Responsible for liaison with the NSSS vendor to resolve interface requirements.

3,000 MWt Reference II Design Study, General Atomic Company
(July 1975-Jan 1976)

As ENGINEER, coordinated the design of the piping and equipment arrangement inside the containment with the goal of reducing HTGR plant costs. The various disciplines coordinated to accomplish this cost reduction included structural, pipe stress, engineered safeguards, and engineering mechanics. The work included development of containment structures; analysis of high energy line break (both for pipe restraint and containment design pressure determination); application of high temperature pipe stress criteria to piping arrangement; arrangement and location of pipe whip restraints.

Responsible for developing pipe sizes for the major steam (main, hot, and cold reheat) and the feedwater systems within the constraints of minimum costs, pipe stress criteria, allowable pressure drops, and maximum fluid velocities.

1200 MWe Nuclear Power Plant, New England Power Company and Central Maine Power Company, Power Plant (July 1974-July 1975)

As ENGINEER, responsible for the development of design criteria and implementation of those criteria for the layout and arrangement of the plant's annulus building. Responsible for the development of design bases, system description, equipment specifications, and PSAR write-ups for several NSSS and reactor auxiliary systems, including chemical and volume control, residual heat removal, boron recovery, liquid waste, gaseous waste, and solid waste. In addition, coordinated the development of the Source Term section of the PSAR and Environmental Report.

<u>PWR SYSTEMS DIVISION AND OFFSHORE POWER SYSTEMS, WESTINGHOUSE ELECTRIC CORPORATION</u> (Aug 1971 - Jun 1974)

As SENIOR SYSTEMS ENGINEER, responsible for design of the reactor plant auxiliary systems (e.g., component cooling water, service water, spent fuel pool cooling and purification, containment leak detection, combustible gas control). Responsibilities included development of design criteria, conformance to design codes, PSAR write-ups, system descriptions, heat balance and fluid flow calculations, and equipment specifications. Supervised the layout and arrangement of assigned systems.

<u>U.S. NAVY - NUCLEAR SUBMARINE FORCE</u> (June 1965-July 1971)

As CHIEF ENGINEER, responsible for the operation and maintenance of the nuclear submarine's propulsion plant. Directed ship's force and coordinated shipyard work during an extensive submarine overhaul. Supervised 4 officers and 35 enlisted men.

ATTACHMENT 4

ROGER F. REEDY, P.E.

Mr. Reedy has worked in the pressure vessel and nuclear power industries since 1956. His experience includes the design, analysis, fabrication, and erection of nuclear power plant components and implementation of the applicable quality systems. His background encompasses boiling water, pressurized water, and HTGR nuclear power plants, as well as pressure vessels and storage tanks for petroleum, chemical, and other energy industries. Mr. Reedy is an acknowledged expert in the design of pressure vessels and nuclear components meeting the requirements of the ASME Boiler and Pressure Vessel Code.

He has been involved in licensing, engineering review, project coordination, and training of personnel. He has testified as an expert witness in litigations and before regulatory groups, including USNRC, ASLB, and ACRS on topics such as design criteria, applications, fabrication techniques, and material applications.

Mr. Reedy has been an active participant for the past 15 years as a member and as chairman of major nuclear Codes and Standards Committees in the development of design, construction and quality criteria for nuclear power plant components. He has served utilities, architect/engineers, and manufacturers as a consultant on all aspects of nuclear power plant licensing, design, quality considerations, and construction.

Roger F. Reedy is currently chairman of the ASME Section III Code for Nuclear Power Plant Components. He is also a member of the N626.3 Committee which developed the rules concerning duties and responsibilities of engineers designing ASME Code components for nuclear plants. This standard specifies minimum qualifications and details the engineer's responsibilities with regard to coordinating material application, fabrication details, quality assurance and non-destructive examinations of the component.

He has worked with the Republic of China Atomic Energy Council to set up an independent quality assurance and inspection program for all nuclear components installed in Taiwan. In addition, for about the past ten years, Mr. Reedy has given lectures on the ASME Code and quality assurance to NRC I & E inspectors in each of the Regions.

Mr. Reedy was one of the initial members of the Pressure Vessel and Piping Division of ASME and helped start the ASME Training Programs for engineers. The program was so successful that other engineering groups have developed similar programs.

Professional Background

- American Society of Mechanical Engineers
  - Boiler and Pressure Vessel Committee
  - Chairman, Subcommittee on Nuclear Power (Section III)
  - Executive Committee, member

  In 1980, he was awarded the 1980 ASME Centennial Medal by the Policy Board for Codes and Standards in recognition of his decades-long contribution to the development of the Boiler and Pressure Vessel Code.

  - Subgroup on Containment, past chairman
  - Subgroup on Fabrication and Examination, former member
  - ASME Pressure Vessel and Piping Division
  - Past Chairman
  - Nuclear Codes and Standards Committee, member
  - ANSI/ASME N626.3 Specialized Professional Engineers Committee, member

Professional Registration

Professional Structural Engineer	Illinois
Professional Civil Engineer	California Illinois Indiana Michigan Wisconsin

Professional Experience

1981 - Present

R.F. REEDY, INCORPORATED
Los Gatos, California
President

Currently consulting with utilities, manufacturers and architect/engineers.

1976 - 1981

NUCLEAR TECHNOLOGY, INCORPORATED
San Jose, California
Successively Manager, Special Projects and Chief Consultant

As Manager, Special Projects, he was responsible for coordinating NUTECH'S quality assurance program and their role as Monitor of the Mark I Containment Modification Project.

His CBI experience and ASME Code (Section III) expertise was a key element in working with the utilities and General Electric to define and execute a modification program acceptable to the U.S. Nuclear Regulatory Commission.

Was then advanced to Chief Consultant, serving as ex-officio advisor to all in-house projects and all clients on design, quality and construction questions concerning application of the ASME Code.

During his term at NUTECH, Mr. Reedy developed and wrote Code Capsule, a biennial commentary on the changes to the ASME Boiler and Pressure Vessel Code.

1956 - 1976

CHICAGO BRIDGE AND IRON COMPANY
Oak Brook, Illinois
Successively Designer, Staff Engineer, Project Engineer, Design Manager and Senior Engineer.

Duties included design of pressure vessels and storage tanks, including cryogenic vessels, vacuum chambers, multi-layer vessels, environmental chambers, and high-pressure chambers. His duties required close liaison with shop and field personnel, providing Mr. Reedy with an intimate knowledge of practical shop and field construction techniques, including the applicable quality requirements.

He has designed more than 50 containment vessels and was the responsible Design Manager for most of the nuclear containment vessels fabricated by CBI. He also designed the first field-erected nuclear reactor.

As Senior Engineer, he consulted with the design staff and other departments concerning ASME Code requirements and special projects.

Education

B.S., Civil Engineering, Illinois Institute of Technology, 1956
Qualified Lead Auditor, ANSI N 45.2.23

AFFIDAVIT OF JOHN M. BIGGS

The undersigned, John M. Biggs, this 12th day of October, 1983, upon his oath states that the attached Curriculum Vitae is a true and correct statement of his education and professional experience.

John M. Biggs

John M. Biggs

October 12, 1983

Debra Burton

Notary Public

DEBRA A. BURTON, Notary Public
My Commission Expires December 30, 1988

ATTACHMENT 1

CURRICULUM VITAE

JOHN M. BIGGS

Education

Massachusetts Institute of Technology

Bachelor of Science in Civil Engineering	1941
Master of Science in Civil Engineering	1947

Professional Employment

Stress Analyst - Curtiss-Wright Corporation	1941-42
Instructor of Civil Engineering - Robert College, Istanbul, Turkey	1942-45
Structural Designer - Fay, Spofford & Thorndike, Consulting Engineers, Boston	1944-49
Massachusetts Institute of Technology	1947-Date
Instructor of Civil Engineering	1947-49
Ass't. Prof. of Civil Engineering	1949-55
Assoc. Prof. of Civil Engineering	1955-63
Professor of Civil Engineering	1963-Date
Emeritus Professor of Civil Engineering	1982-Date
Director, Civil Engineering Systems Laboratory	1964-67
Acting Head, Structures Division	1967-68
Head, Structures Group	1976-82
Partner, Hansen, Holley and Biggs, Consulting Engineers, Cambridge, MA	1955-80
Director, Hansen, Holley and Biggs, Inc. Cambridge, MA	1975-Date

Professional Societies, etc.

Registered Professional Engineer, Commonwealth of MA	
Member, American Society of Civil Engineers	
Chairman, Structural Division	1970-71
Executive Committee, Structural Division	1968-72
Chairman, Committee on Wind Forces	1955-60
Member, Committee on Electronic Computation	1957-60
Member, Committee on Plasticity Related to Design	1957-59
Member, Committee on the Limitations of Bridge Deflection	1956-61
Member, Administrative Committee on Loads and Stresses	1955-60
Member, Committee on Lifeline Earthquake Engineering	1973-74
Member, Boston Society of Civil Engineers	
Chairman, Structural Section	1957-58
Director	1959-61
Vice-President	1964-67
President	1966-67

Member, Column Research Council 1958-72
Member, Mayor's Committee on the Revision of the Boston Building Code

Subcommittee on Steel Design 1956-63
Subcommittee on Loads 1958-60

Member Advisory Committee, Massachusetts State Building Code 1974-Date
Member, Advisory Panel on Bridges, AASHO Road Test
Member, Committee on Bridges, Highway Research Board, National Academy of Sciences 1961-65
Member, Committee on Design Loads for Buildings, American Standards Association 1962-66

Recent Publications

"Introduction to Structural Dynamics," McGraw-Hill Book Co., NY, 1964

"Structural Response to Seismic Input," page 306, Seismic Design for Nuclear Power Plants, MIT Press, Cambridge, MA, 1970

"Seismic Analysis of Equipment Mounted on a Massive Structure," page 319, Seismic Design for Nuclear Power Plants, MIT Press, Cambridge, MA, 1970 (with J. Roesset).

"Computer System for the Analysis and Design of Reinforced Concrete Structures," ACI Journal, April, 1970 (with P.J. Pahl and H.N. Wenke).

"Soil-Structure Interaction in Nuclear Power Plants," 3rd Japanese Symposium on Earthquake Engineering, Tokyo, November, 1970 (with R.V. Whitman).

"Integrated System for RC Building Design," Journal of the Structural Division, ASCE, Vol. 97, January, 1971 (with P.J. Pahl and H.N. Wenke).

"Earthquake Code Evolution and the Effect of Seismic Design on the Cost of Buildings," MIT Department of Civil Engineering, Report No. R72-20, May, 1972 (with S.J. Leslie).

"Seismic Response Spectra for Equipment in Nuclear Power Plants," Proceedings, First International Conference on Structural Mechanics in Reactor Technology, Berlin, July, 1972.

"Parametric Analysis of Soil-Structure Interaction for a Reactor Building," Proceedings, First International Conference on Structural Mechanics in Reactor Technology, Berlin, July, 1972 (with J.T. Christian and R.V. Whitman).

"Seismic Response of Buildings Designed by Code for Different Earthquake Intensities," MIT Department of Civil Engineering, Report No. R73-7, January, 1973 (with P.H. Grace).

"Seismic Design Decision Analysis," Journal of the Structural Division, ASCE, Vol. 101, ST5, May, 1975 (with R.V. Whitman, et al).

"Comparison of Seismic Analysis Procedures for Elastic Multi-Degree Systems," MIT Department of Civil Engineering, Report No. R76-5, January, 1976 (with E.H. Vanmarcke et al.).

"Variability of Inelastic Structural Response Due to Real and Artificial Ground Motions," MIT Department of Civil Engineering, Report R76-6, January, 1976 (with E.H. Vanmarcke, Robert A. Frank, et al.).

"Studies on the Inelastic Dynamic Analysis and Design of Multi-Story Frames," MIT Department of Civil Engineering, Report R76-29, July, 1976 (with W.H. Luyties and S.A. Anagnostopoulos).

"Inelastic Response Spectrum Design Procedures for Steel Frames," MIT Department of Civil Engineering, Report No. R76-40, September, 1976 (with Richard W. Haviland).

"On the Safety Provided by Alternate Seismic Design Methods," MIT Civil Engineering Department, Report No. R77-22, July, 1977 (with D.A. Gasparini).

"Inelastic Dynamic Design of Steel Frames to Resist Seismic Loads," MIT Civil Engineering Department, Report No. R77-23, July, 1977 (with J.H. Robinson, Jr.).

"Use of Inelastic Spectra in Aseismic Design," Journal of the Structural Division, ASCE, Vol. 104, No. ST1, January 1978 (with S.A. Anagnostopoulos and R.W. Haviland).

"Inelastic Response Spectra for Aseismic Building Design," Journal of the Structural Division, ASCE, Vol. 106, No. ST6, June, 1980 (with S.P. Lai).

"Seismic Effectiveness of Tuned Mass Dampers," Journal of the Structural Division, ASCE, Vol. 107, No. ST8, August, 1981 (with A.M. Kaynia and D. Veneziano).

"Seismic Damage in Reinforced Concrete Frames," Journal of the Structural Division, ASCE, Vol. 107, No. ST9, September, 1981 (with H. Banon and H.M. Irvine).

"Flexible Sleeved-Pile Foundations for Aseismic Design," MIT Civil Engineering Department, Report No. R82-04, March, 1982.

AFFIDAVIT OF MYLE J. HOLLEY, JR.

The undersigned, Myle J. Holley, Jr., this 12th day of October, 1983, upon his oath states that the attached statement of Professional Experience is a true and correct statement of his education and professional experience.

Myle J. Holley, Jr.

Myle J. Holley, Jr.

October 12, 1983

William S. Moonan

Notary Public

WILLIAM S. MOONAN
NOTARY PUBLIC
MY COMMISSION EXPIRES
AUGUST 6, 1987

ATTACHMENT 2

STATEMENT OF PROFESSIONAL EXPERIENCE

MYLE J. HOLLEY, JR.

Mr. Holley received the S.B. and S.M degrees in Civil Engineering from MIT in 1939 and 1947, respectively. From 1939 to 1946 he was employed by the S. Morgan Smith Co. (now the York, PA Division of Allis-Chalmers Manufacturing Co.) as a stress analyst and designer of heavy machinery. In 1946 he joined the Faculty of MIT in the Department of Civil Engineering. While on that faculty he taught subjects in structural analysis and design, and supervised structural research projects. The latter included work in the fields of massive reinforced concrete structures, prestressed concrete, structural applications of granite, high-strength reinforced concrete beams, and the performance of thin arch concrete dams. For several of his years on the MIT Faculty, Mr. Holley was in charge of the Structural Division of the Civil Engineering Department.

In 1955 Professor Holley and his colleagues, Professors John M. Biggs and Robert J. Hansen, formed the consulting partnership Hansen, Holley and Biggs. Since 1975 the group has functioned as Hansen, Holley and Biggs, Inc. Mr. Holley's participation in the professional efforts of the group has continued undiminished since his retirement from teaching in 1974.

The professional assignments of Hansen, Holley and Biggs have been related, almost exclusively, to complex problems of structural design and structural behavior. Their clients include both engineering firms and owners of major constructed facilities. A substantial fraction of their practice has involved advice and assistance in the resolution of problems arising in the design and construction of nuclear power plants. In this area of their practice, clients have included:

Stone and Webster Engineering Corporation
United Engineers and Constructors
Gibbs and Hill
American Electric Power Corporation
Rochester Gas and Electric Company
Portland Gas and Electric Company

Mr. Holley has been extensively involved in structural aspects of nuclear power plant projects for all of the above companies. In addition, he has been a consulting member of several internal design review boards conducted by Stone and Webster Engineering Corporation.

Mr. Holley is a registered Professional Engineer in the Commonwealth of Massachusetts. He is a member of the American Society of Civil Engineers, the American Concrete Institute, and the American Society for Engineering Education. He has served on numerous professional committees. This has included several years on ACI 349 Concrete Nuclear Structures and ACI 359 Nuclear Reactor Components, and he currently is a consulting member of these committees.

AFFIDAVIT OF RONALD WRAY

The undersigned, Ronald Wray, this 12th day of October, 1983, upon his oath states that the attached Professional Resume is a true and correct statement of his education and professional experience.

Ronald Wray

Ronald Wray

October 12, 1983

William S Moonan

Notary Public

WILLIAM S. MOONAN
NOTARY PUBLIC
MY COMMISSION EXPIRES
AUGUST 6, 1987

TELEDYNE
ENGINEERING SERVICES

RONALD WRAY
Manager, Engineering
Analysis

Professional Resume

Education

Northeastern University, B.S. in Civil Engineering, 1956
Rensselaer Polytechnic Institute, M.S. in Engineering Science, 1962

Experience

Teledyne Engineering Services, and Teledyne Materials Research, since 1971: theoretical stress analysis of pressure vessels, piping systems and frame structures utilizing computer program solutions and finite element methods; performed and directed static and dynamic analyses of Nuclear and LNG Piping Systems; conducted design reviews of Nuclear Piping Systems.

Instructor at Franklin Institute of Boston, Evening Division

AVCO Systems Division, 1962-1971: performed detailed stress and buckling analysis of various reentry vehicle shell structures under combined reentry pressure and inertia loads and heating. Designed and analyzed large vacuum and pressurized chambers for a portable sterilization/clean room facility built for NASA/Langley; responsible for the structural design and evaluation of space power systems and planetary probe systems.

Pratt & Whitney Aircraft, Canal Division, 1958-1962: performed and directed detailed analyses and design evaluation of nuclear reactor core components and pressure vessels; conducted thermo-structural analysis of system piping and heat exchangers involving liquid metal coolants under conditions of high temperature operation an severe transients; established design criteria for components exposed to long-life, high-temperature conditions,

U.S. Army Corps of Engineers, 1st Lieutenant, 1956-1958: served as project officer on military construction sites; field experience in reinforced concrete an steel erection.

Membership

ASME, Boiler and Pressure Vessel Code, Chairman, Special Working Group on Dynamic Analysis.

1/82

IDVP Exhibit List

(1) Independent Design Verification Program Final Report, Diablo Canyon Nuclear Power Plant - Unit 1 (as revised through 10/10/83).

Purpose: Summary of the IDVP efforts and statement of conclusions and evaluations of the IDVP.

Sponsoring witness: Dr. William E. Cooper

(2)(a) Diablo Canyon Nuclear Power Plant Design Verification Program Management Plan, Phase I (March 29, 1982)

(b) Diablo Canyon Nuclear Power Plant Design Verification Program Management Plan, Phase II (June 18, 1982)

Purpose: Description of program plans of the IDVP.

Sponsoring witness: Dr. William E. Cooper

(3) Interim Technical Reports (ITR's) of IDVP, as listed in Attachment A hereto.

Purpose: Depending upon the ITR, documentation of programmatic aspects of the IDVP or report of detailed technical results.

Sponsoring witness: Dr. William E. Cooper for ITR's issued by TES
Dr. Robert L. Cloud for ITR's issued by RLCA
Mr. John E. Krechting for ITR's issued by SWEC
Mr. Roger F. Reedy for ITR's issued by RFR

ITR	REV NO.	ISSUE DATE	ISSUED BY	TITLE
1	1	821022	RLCA	Additional Verification and Additional Sampling (Phase 1)
2	0	820623	TES	Comments on the R.F. Reedy, Inc., Quality Assurance Audit Report on Safety-Related Activities Performed by PGandE Prior to June 1978
3	0	820716	RLCA	Tanks
4	0	820723	RLCA	Shake Table Testing
5	0	820819	RLCA	Design Chain
6	0	820910	RLCA	Auxiliary Building
7	0	820917	RLCA	Electrical Raceway Supports
8	0	821005	RLCA	Independent Design Verification Program for Verification of PGandE Corrective Action
9	0	821015	RFR	Development of the Service-Related Contractor List for Non-Seismic Design Work Performed for DCNPP-1 Prior to June 1, 1978
10	0	821029	RLCA	Verification of Design Analysis Hosgri Spectra
11	0	821102	TES	PGandE-Westinghouse Seismic Interface Review
12	0	821105	RLCA	Piping
13	0	821105	RLCA	Soils - Intake Structure
14	2	830725	SWEC	Verification of the Pressure, Temperature, Humidity, and Submergence Environments used for Safety-Related Equipment Specifications Outside Containment for Auxiliary Feedwater System and CRVP System

ITR	REV NO.	ISSUE DATE	ISSUED BY	TITLE
15	0	821210	RLCA	HVAC Duct and Supports Report
16	0	821208	RLCA	Soils - Outdoor Water Storage Tanks
17	0	821214	RLCA	Piping - Additional Samples
18	1	830524	SWEC	Verification of the Fire Protection Provided for Auxiliary Feedwater System Control Room Ventilation and Pressurization System Safety-Related Portion of the 4160V Electric System
19	0	821216	SWEC	Verification of the Post-LOCA Portion of the Radiation Environments used for Safety-Related Equipment Specification Outside Containment for Auxiliary Feedwater System and Control Room Ventilation and Pressurization System
20	2	830725	SWEC	Verification of the Mechanical/Nuclear Design of the Control Room Ventilation and Pressurization System
21	1	830503	SWEC	Verification of the Effects of High Energy Line Cracks and Moderate Energy Line Breaks for Auxiliary Feedwater System and Control Room Ventilation and Pressurization System
22	2	830725	SWEC	Verification of the Mechanical/Nuclear Portion of the Auxiliary Feedwater System
23	1	830527	SWEC	Verification of High Energy Line Break and Internally Generated Missile Review Outside Containment for Auxiliary Feedwater System and Control Room Ventilation and Pressurization System

ITR	REV NO.	ISSUE DATE	ISSUED BY	TITLE
24	1	830504	SWEC	Verification of the 4160V Safety-Related Electrical Distribution System
25	1	830429	SWEC	Verification of the Auxiliary Feedwater System Electrical Design
26	1	830502	SWEC	Verification of the Control Room Ventilation and Pressurization System Electrical Design
27	2	830725	SWEC	Verification of the Instrument and Control Design of the Auxiliary Feedwater System
28	2	830725	SWEC	Verification of the Instrument and Control Design of the Control Room Ventilation and Pressurization System
29	0	820117	SWEC	Design Chain - Initial Samples
30	0	830112	RLCA	Small Bore Piping Report
31	1	830804	RLCA	HVAC Components
32	1	830401	RLCA	Pumps
33	1	830428	RLCA	Electrical Equipment Analysis
34	1	83[illegible]24	SWEC	Independent Design Verification of DCP Efforts by SWEC
35	0	830401	RLCA	Independent Design Verification Program Verification Plan for DCP Activities
36	1	830620	SWEC	Final Report on Construction Quality Assurance Evaluation of G.F. Atkinson
37	0	830223	RLCA	Valves
38	2	830620	SWEC	Final Report on Construction Quality Assurance Evaluation of Wismer and Becker
39	0	830225	RLCA	Soils - Intake Structure Bearing Capacity and Lateral Earth Pressure

ITR	REV NO.	ISSUE DATE	ISSUED BY	TITLE
40	0	830309	RLCA	Soils Report - Intake Sliding Resistance
41	0	830419	RFR	Corrective Action Program and Design Office Verification
42	0	830415	RFR	R.F. Reedy, Inc., Independent Design Verification Program Phase II Review and Audit of PGandE and Design Consultants for DCNPP-1
43	0	830414	RLCA	Heat Exchangers
44	0	830415	RLCA	Shake Table Test Mounting Class 1E Electrical Equipment
45	0	830517	SWEC	Additional Verification of Redundancy of Equipment and Power Supplies in Shared Safety-Related Systems
46	0	830627	SWEC	Additional Verification of Selection of System Design Pressure and Temperature and Differential Pressure Across Power-Operated Valves
47	0	830627	SWEC	Additional Verification of Environmental Consequences of Postulated Pipe Ruptures Outside of Containment
48	0	830727	SWEC	Additional Verification of Jet Impingement Effects of Postulated Pipe Ruptures Inside Containment
49	0	830623	SWEC	Additional Verification of Circuit Separation and Single Failure Review of Safety-Related Electrical Equipment
50	0	830722	TES	Containment Annulus Structure Vertical Seismic Evaluation
51	1	830915	TES	Containment Annulus Structure - Verification of DCP Corrective Action

ITR	REV NO.	ISSUE DATE	ISSUED BY	TITLE
54	1	831003	RLCA	Corrective Action Containment Building
55	1	831001	RLCA	Corrective Action Auxiliary Building
56	1	830924	RLCA	Corrective Action Turbine Building
57	1	830908	RLCA	Review of DCP Activities Fuel Handling Building
58	1	831001	RLCA	Verification of DCP Activities Intake Structure
59	1	830924	RLCA	Corrective Action Large Bore Piping
60	1	831003	RLCA	Corrective Action Large and Small Bore Pipe Supports
61	1	831002	RLCA	Corrective Action Small Bore Piping
63	1	831002	RLCA	Corrective Action HVAC Ducts, Raceways, Instrument Tubing and Associated Supports
65	1	831010	RLCA	Corrective Action Rupture Restraints
67	1	830909	RLCA	Corrective Action, Equipment
68	1	831004	RLCA	Verification of HLA Soils Work

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that copies of the letter from Maurice Axelrad to the Appeal Board dated October 14, 1983, and its enclosures (pre-filed direct testimony of three panels of IDVP witnesses, qualifications of IDVP witnesses, and IDVP exhibit list) have been served on the following by deposit in the United States mail, first class, postage pre-paid, this 14th day of October, 1983, except that, in the case of individuals designated by an asterisk, arrangements have been made for delivery by courier or personal delivery no later than 10:00 a.m. on October 17th:

*Dr. John H. Buck
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U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

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Maurice Axelrad