AFFIDAVIT

My name is Richard D. Parks. I work as an investigator for the Government Accountability Project (Gap) in the ongoing investigation of the Diablo Canyon Power Plant (DCPP). My review of publically available documents leads me to believe that Pacific Gas and Electric (PG&E) is subject to preferential treatment by the NRC, which has granted dispensation to or in some instances ignored the requirements and recommendations contained in the following documents:

1. NUREG-0737
2. NUREG-0588
3. NUREG-0531
4. NUREG-0691
5. ~~NUREG-0817 (Appendix A to license # DPR-76)~~ RP
6. ANSI B31.1
7. REG Guide 1.143

Items #1 and #2 above are significant because, they require extensive backfitting/upgrading of the nuclear safety related equipment in use at DCPP to minimize the chances of or mitigate the consequences of an accident similar to the one that occurred at TMI-2.

Items #3 and #4 above are significant because they identify and recommend corrective action pertaining to corrosion and cracking of sensitized stainless steel. The potential for this exists at the DCPP, due to the materials involved and allegations of deficient material control.

Item #5 above is significant because it contains the requirement for the establishment, approval and implementation of a Process Control Program for administering

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radioactive waste processing at DCPP.

Items #6 and #7 are significant because they establish the requirements that must be adhered to in the design, fabrication and testing of radioactive waste handling systems for nuclear power plants.

Problem #1 (items 1,2): Lack of compliance with TMI related NUREGS

NUREG-0737;"Clarification of TMI Action Plan Requirements" submitted November 1980, includes a letter from D.G. Eisenhut to "All Licensees of Operating Plants and Applicants For Operating Licenses and Holders of Construction Permits" In the fourth paragraph of the letter Mr. Eisenhut states "The requirements herein...are applicable to applicants for operating licenses and such applicants are expected to meet the same schedule of implementation as indicated for operating reactors...any item for which the implementation 's date is prior to the expected date of issuance of an operating license will be considered to be a prerequisite to obtaining that license."

A review of license amendments and requests by PG&E available in the Cal Poly Library, the local Public Document Room (PDR), reveals the following NUREG-0737 requirements with which PG&E may not be in full compliance:

Requirement A. NUREG item IIB.3 "Post Accident Sampling"
license condition 2.C(8)h
Requirements issued 9/13/71
Interim System required by 1/1/80
Plant Modifications required by 1/1/82

Under "Documentation Required" II.B.3 (enclosure 3 to NUREG-0737) Operating license applicants must..."provide a

description of the implementation of the position and clarification including P&ID 's, together with either (a) a summary description of procedures for or (b) copies of procedures for...in accordance with the proposed review schedule but in no case less than 4 months prior to the issuance of an operating license." PG&E letter 11/11/83, "Updated status of compliance with license conditions 2.C(8)h, 2C(8)k and 2.C(8)l(2) states that PG&E is relying on the interim Post-LOCA (Loss of Coolant Accident) Sampling System and revisions to "interim procedure for estimating core damage (CAP-G-4). This sampling program is necessary to avoid mistakes and investigate the consequences if an accident occurred. The authorization for this "interim compliance" was issued by NRC letter (D.G. Eisenhut to J.O. Schuyler) dated Nov.15, 1983. From the public record available at the PDR, the degree to which this requirement has been sacrificed remains indeterminate.

Requirement B. NUREG item II.K.3.30 and II.K.3.31 "Calculations for Small Break LOCA 's"

license condition 2.C(8)O
requirements issued 5/1/80
implementation required 1/1/83
Under "Documentation Required" II.K.3.30 (enclosure 3 to NUREG-0737) four specifics are addressed:

1) Licensees shall submit outline of program for model justification/revision by 11/15/80.

2) Licensees shall submit additional information for model justification and/or revised analysis model for staff approval by 1/1/82

3) Licensees shall submit their plant-specific analysis using the revised models by 1/1/83 or one year after any model revisions are approved.

4) Applicants shall submit appropriate information in accordance with the licensing review schedule.

Compliance with this requirement is necessary to demonstrate that each specific plant as designed is capable of withstanding a TMI-type accident. It also demonstrates that the operators can maintain the plant in a safe condition by properly diagnosing the symptoms and responding.

Under "Documentation Required" II.K.3.31 (enclosure 3 to NUREG-0737) the requirements for "Operating License Applicants" states

1) All applicants...should submit documentation 4 months prior to the expected issuance of the staff safety evaluation report for an operating license or 4 months prior to the listed implementation date, whichever is later.

The authorization for "relief" from this requirement was granted by NRC letter (D.G. Eisenhut to J.O. Schuyler, dated Nov. 15, 1983) which states: "...the NRC staff is treating this (requirement) on a generic basis. The staff is currently reviewing the Westinghouse Corporation generic submittal (NUREG II.K.3.30)...We require that the PG&E company submit it 's plant specific analysis (NUREG-II.K.3.31) which must be approved by the NRC staff within one year from the date of NRC approval of the Westinghouse generic models."

Requirement C) NUREG item II.B.2 "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in POSTACCIDENT OPERATIONS"

license condition 2.C.(5)
requirements issued 9/13/79
review designs by 1/1/80

plant modifications by1/1/82
Equipment Qualification by 6/30/82

Compliance is necessary to demonstrate that certain essential safety-related equipment can still function under the extreme conditions following an accident.

Under "Changes to Previous Requirements and Guidance II.B.2 (enclosure 3 to NUREG-0737) item #6 states:"Because of difficulty in obtaining equipment (eg, remote-operated valves), the implementation date is moved to 1/1/82, or the first outage of sufficient duration thereafter, but no later than July 1, 1982.

The Environment Qualification Requirements were further delineated in section 7.8 Safety Evaluation Report Supplement No. 9 (requiring compliance with NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment) which required that:

"a No later than June 30, 1982 PG&E shall be in compliance with the provisions of NUREG-0588, "Interim Staff position on Environmental Qualification of Safety-Related Electrical Equipment", for safety related equipment exposed to harsh environment.

b) Complete and auditable records must be available and maintained at a central location which describes the environmental qualification method for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG_0588. Such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified to document complete compliance by June 30, 1982.

c) The licensee shall provide affirmation of implementation or the surveillance and maintenance program procedures prior to the issuance of a full power license, and adhere to the commitments of their

September 2, 1981 submittal which will result in compliance with NUREG-0588."

In a PG&E letter dated June 23, 1982 PG&E requested "relief" from the implementation required date of NUREG-0737 and NUREG-0588 and requested extension of required implementation date to the second refueling outage. PG&E stated that they would comply with the requirements and schedule of item (c) above. The NRC responded favorably to PG&E 's request for relief via letter dated June 30, 1982 from F.J. Moraglia, Chief Licensing Branch #3 to M.H. Furbrush.

I challenge the basis for this relief, due to the allegations whistleblowers have disclosed to me over the last month. Witnesses have disclosed documentation that demonstrates violations of even internal environmental qualification procedures. Safety-related valves were disassembled and repaired without adequate documentation, leading to possible degradation of the components. See exhibit 1 for a report of my research and relevant documentation.

Requirement D) NUREG item III.D.1.1 "Integrity of Systems Outside Containment Likely to Contain Radioactive Material For Pressurized-Water Reactors and Boiling Water Reactors".

requirements issued 9/29/79
implementation required prior to full power

Clarification: (Enclosure 3 to NUREG-0737) states:"Applicant shall provide a summary description, together with initial leak-test results, of their program to reduce leakage from systems outside containment that would or could contain primary coolant or other highly radioactive fluids or gases during or following a serious transient or accident.

1) Systems that should be leak tested are as follows (any other plant system which has similar functions or

postaccident characteristics even though not specified herein, should be included):

a. Residual Heat Removal (RHR)
b. Containment Spray Recirculation
c. High-pressure Injection Recirculation
d. Containment and Primary Coolant Sampling
e. Reactor Core Isolation Cooling
f. Make-up and Letdown (PWR only)
g. Waste Gas (includes headers and cover gas system outside of containment in addition to decay or storage system) include a list of systems containing radioactive materials which are excluded from the program and provide justification for exclusion.

2) Testing of gaseous systems should include Helium leak detection or equivalent testing methods.

3) Should consider program to reduce leakage potential release paths due to design and operator deficiencies..."

Implementation: (Enclosure 3 to NUREG-0737)
This requirement shall be implemented by applicants for operating license prior to issuance of a full power license.

Documentation: (Enclosure 3 to NUREG-0737)
Applicants shall submit the information requested in the "clarification" section of this position at least 4 months prior to issuance of a fuel-loading license.

Compliance is necessary to maintain isolation between Unit I and Unit II. Otherwise radiation from Unit I could contaminate Unit II and it 's workers, during an accident or potentially from a low-level release during operations. A review of publicly available documents/records has not revealed a PG&E submittal of the required program or the NRC approval of the same. As a result, there is still no demonstrated assurance that when Unit I begins full-power operation it will be isolated from Unit II. This requirement is too significant to waive or postpone.

Problem #2 (items 3,4): Lack of implementation of

Recommendations Stemming From NUREG-0531 and NUREG-0691.

Requirement A. NUREG-0531 "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants" issued February 1979.

This NUREG was the product of a Pipe Cracking Study Group formed to investigate the phenomena of "Intergranular Stress-Corrosion Cracking" (IGSCC) in boiling water reactor piping. However not all their findings, conclusions or recommendations were restricted to BWR 's. Particular interest should be directed to the following sections due to the direct impact the statements have on DCPP.

Item 1: RESPONSE TO CHARTER QUESTION #2 page XIV Question2: "Resolution of concerns raised over the ability to use Ultrasonic technique to detect cracks in austenitic stainless steel."

"If present code evaluation standards are used, many cases of IGSCC will not be properly identified..."

"It is the study group 's opinion that ultrasonic examination can be effective in identifying most IGSCC before leaks occur if the most susceptible welds are examined at frequent intervals, if equipment especially suited to detect IGSCC is used, and if improved evaluation methods are used."

The significance of this statement is simply that the type of material used at DCPP (316 stainless steel) in the reactor coolant system is susceptible to IGSCC and therefore requires special detection capability. A review of Board Notification Letters No. 83-96 and 83-112 (both deal with the investigation into Reactor Coolant System Piping minimum wall thickness violations) already identified UT methods are incapable of determining pipe wall thickness because of the inherent nature of stainless steel piping(Para 8.3.1 NUREG-0531). These reports

would then also question whether the UT methods employed by PG&E could detect IGSCC in the pipe wall because the materials in use in the Reactor Coolant System at Diablo Canyon is 316 type stainless steel. NUREG-0531 addresses IGSCC in type 304, 316 stainless steel.

Item 2: Response to Charter Question 4 (pages XV and XVI) Question 4: "The Potential for Stress Corrosion Cracking in PWRS."

Under the paragraph titled "Systems other than Primary Systems" it is stated, "These incidences of stress-corrosion cracking have generally occurred in the heat-affected zones of welds in austenitic stainless-steel pipe, but they have also been reported in base metal that was sensitized.... We believe that the NRC has initiated proper action to define he problem and initiate industry efforts to control it. The proposed corrective action defined by licensees should be reviewed by the NRC staff and appropriate action taken to assure satisfactory resolution of this matter."

The significance of these statements is simply that a problem exists in the "non-primary(NSSS) systems" and the licensee should develop a corrective action plan to be reviewed by the NRC. However, a review of publicly available documents/records has not disclosed the existence of such a plan or the NRC 's approval of such a plan.

The safety significance of this statement is that IGSCC induced failure of various plant systems may degrade the function of the systems below that assured in the FSAR for certain postulated conditions.

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Item 3: "PWR Cracking Experience and Corrective Actions (page 3.1) Paragraph 3.1 "...several facilities have furnace-sensitized safe ends. Most facilities that had furnace-sensitized safe ends implemented a repair program to apply protective cladding. However, a few PWRS still have furnace-sensitized safe ends. The PWR facilities that have furnace-sensitized safe ends include San Onofre 1, Haddam Neck, H. B. Robinson 2, and Diablo Canyon 1. In-service inspections on these safe ends are required."

The significance of this statement is simply that PG&E was aware in 1979 that the safe-ends (a method employed to attach stainless steel pipe to carbon steel vessel) in use at Unit 1 was susceptible to stress-corrosion cracking. PG&E had ample time to perform a repair program. One simple reason the other three listed plants had not was because they are operational. However, DCPP-1 was not. A review of publicly available records has not disclosed a corrective action program for this deficiency. For reasons previously discussed, doubt exists as to whether or not PG&E UT methods can detect IGSCC/SCC in their furnace-sensitized safe-ends.

Item 4: para. 4.3.1 "Effect of Composition" (page 4.4) This paragraph states in part, "The most significant factor affecting the degree of sensitization is the carbon content of the alloy. ...Low carbon- grade stainless steels (.03% max) have significantly lower susceptibility than do regular grade stainless steels (.08% max carbon),..."

The significance of this statement is self-evident. I have been informed that the stainless steel carbon content at Diablo Canyon varies between .06 and .07% and thus is very susceptible to IGSCC/SCC. If this is true PG&E should determine what systems are involved and incorporate a corrective action program.

Item 5 para. 6.3 "Fabrication Stresses (page 6.4) This paragraph states in part, "The most significant fabrication stresses in connection with IGSCC are probably due to welding and rough grinding without annealing."

Paragraph 6.7 Recommendations states in part....

1. "It is recommended that work be continued to qualify residual stresses that are due to welding....

4. It is recommended that grinding of the interior surface of well regions be avoided under conditions where IGSCC maybe present."

The significance of this statement is simply that welding and grinding on material susceptible to IGSCC has to be strictly controlled to preclude providing the stress ingredient for IGSCC.

A review of the following documents casts doubt as to whether these processes were strictly controlled at DCPP-1.

a. Regulatory Operations Report no. 73-03 dated 6-4-73 identified the following:

1. rusting stainless steel welds not conforming to specifications

2. discrepant stainless steel pipe spools not identified as discrepant

3. post-welded heat treatment to welded stainless steel pipe (after fabrication) to achieve a fit of flanged spool pieces.

b. Regulatory Operations Report no. 73-05 dated 10-15-73 identified the following:

1. PG&E had issued a stop work order (9-9-73) against Wismer-Becker to stop all welding on RCS piping for the following reasons:

a. preferential sequence welding for alignment purposes introduced unusual

stresses
b. "Block" welding was being performed contrary to specifications
c. weld repairs (air-arc gouging method) were performed without qualified written repair procedures

Note: PG&E later "grandfathered" the governing procedures to allow the type of welding and weld repairs when deemed necessary.

c. Regulatory Operations Report no. 74-02 dated 3-4-74 identified the following:

1. while observing the final weld pass on weld joint 3-5B, it was determined that the Reactor Coolant Pump was being drawn out of tolerance by .004 inches beyond the .120 inch tolerance on misalignment allowed by Westinghouse.

PG&E letter dated 2-14-74 issued their final review decision on the problems identified in September 1973. Their conclusion was "acceptable". RO inspection report 374-02 dated 3-4-74 states that "on 1-3-74 PG&E lifted entirely the order of Sept. 20, 1973 allowing unrestricted welding.." The only lack of compliance identified (on the out of alignment tolerance on RCP #3) was documented on a Deviation Report. The NRC had apparently accepted PG&E 's engineering disposition of "acceptable".

d. "Report on Investigation of Reactor Coolant Pipe Weld Thickness at Diablo Canyon" dated 7-1-83, attachment to Board Notification no. 83-112. A review of this document identifies the following:

page 11-3, para.c: History and Controls on Grinding
This paragraph and associated subparagraphs identify that the RCS welds were ground on the inside and outside preparation of surfaces for Pre-Service Inspection.

In conclusion, the methods of welding and grinding performed on the Reactor Coolant System do not demonstrate that the relevant stresses were avoided. PG&E apparently did not recognize the significance of this statement. They had no problem with the welding other than the potential for minimum wall thickness wall violations and totally ignored the implication of inducing the stresses required for IGSCC.

Requirement B. NUREG-0691 "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors" issued in September 1980.

This NUREG was the product of a Pipe Cracking Study Group formed to investigate Stress Corrosion Cracking (SCC) in PWR 's. The study included a generic description for comparison to selected PWR systems at a specific plant (Beaver Valley Power Station - Unit 1) whose systems represent a typical Westinghouse PWR.

> Item 1: Response to Charter Question 1 (page 3)
>
> The response states in part "The results of simplified scoping analysis performed by the PCSG indicate that small line breaks in low head safety injection system...or in the function of the systems below that assumed in the FSAR for certain postulated conditions.
>
> Based on the results of the generic scoping analysis, the PCSG believes that further plant-specific scoping and more detailed analyses should be conducted to better define the safety implications of small line breaks."
>
> The significance of the above statement is self explanatory. SCC induced failure could limit the capacity of various safety related systems to perform as designed.
>
> Item 2: Response to Charter Question 2 (page 7)
>
> The response states in part "The effectiveness for current ISI is considered marginal for PWR secondary systems...."
>
> Item 3: Response to Charter Question 3 (page 10)
>
> The response states in part "Plant-specific analysis should be performed for all plants, including those at the CP stage, to identify PWR secondary systems whose function may be significantly degraded by small-line breaks...."

A review of documents publicly available has not revealed PG&E implementation of recommendations of NUREG-0531 or NUREG-0691. These recommendations, if implemented and combined with the satisfactory implementation and approval of NUREG item IIK.3.31 (calculations for small break LOCA 's), would assure an adequate margin of safety with respect to line break degradation

of safety-related systems.

~~Problem #3 (item 5) "Lack of Process Control Program"~~ Section 6.13 of NUREG-0817 (Appendix A to license #DPR-76 - Diablo Canyon) states that "The PCP shall be approved by the Commission prior to implementation."

A review of publically available documents in the PDR at CAl-Poly library has not disclosed the existence or NRC approval of the process which is the administrative system for radioactive waste handling. This program should be reviewed, ~~approved and made publically available prior to full power licensing.~~ RP

Problem #4 (items 6 and 7) "Lack of Compliance with REG GUIDE 1.143 and ANSI B31.1. REG Guide 1.143 "Design Guidance for Radioactive waste management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

Contains in part: paragraph c. "Regulatory Position" - "The systems should be designed and tested to requirements set forth in the codes and standards listed in Table 1 supplemented by regulatory positions 1.1.2. and 4 of this guide."

Table 1 - "Equipment Codes" requires that all piping and valves be designed and fabricated to ANSI B31.1, Welding procedures qualified to ASME Code Section IX with Inspection and Testing to the requirements of ANSI B31.1.

Paragraph 6 "Quality Assurance for Radwaste Management Systems" defines the QA program acceptable for the NRC staff....

> 4.2.3 Quality Control. The design, procurement, fabrication and construction activities shall conform to the Quality Control provisions of the Codes and standards specified herein. In addition, or where not covered by the following quality control features shall be established.

ANSI B31.1 1977 Edition, Chapter VI "Examination Inspection and Testing" includes paragraph 136.4 "Examination of Welds". The intent of this paragraph is to establish the QC methods for determining the acceptability of welding performed pursuant to this code.

PG&E has classified their radwaste systems as Class 'E' systems and as such the systems are not subject to QC inspection.

In summary, the accident at TMI-2 was the catalyst that enacted various TMI-related NUREGs. These NUREGs were intended to make the Nuclear Plants in this country safer. It is extremely important to fully implement these requirements at DCPP-1 prior to full power licensing. Due to the proximity of the site of DCPP-1 with respect to the HOSGRI fault, the "defense-in-depth" approach should be enforced, instead of minimum compliance with the requirements.

Allowing the plant to go critical, and subsequently licensing for full power, with systems known to be prone to failure due to IGSCC/SCC is unexplainable. A failure once in operation will place an unwarranted financial burden on the rate-payers due to corrective costs and purchase power costs, not to speak of the possible risk to public safety.

The one lesson that TMI should have taught the industry and the NRC is that "NUKE PLANTS" have to be safe - no matter what. I don't believe the NRC has learned the lessons TMI taught us. Another accident similar to TMI would not only be the death of the industry but would have adverse impact on California Economy.

I have read the above 16 page document and it is true and accurate to the best of my knowledge.

Richard D Parks

Richard D. Parks

Subscribed and sworn to before me this 5 th day of ~~May~~ June RP, 1984.

Lisa R Wenter

Notary Public in and for the County of San Luis Obispo, State of California

OFFICIAL SEAL
LISA R. WENTER
NOTARY PUBLIC - CALIFORNIA
SAN LUIS OBISPO COUNTY
My Comm. Expires May 9, 1986