

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

PACIFIC GAS AND ELECTRIC COMPANY

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

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}
}
Docket Nos. 50-275 OL
50-323 OL

AFFIDAVIT OF CHU-YU LIANG
REGARDING RESIDUAL HEAT REMOVAL SYSTEM

I, Chu-yu Liang, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Senior Nuclear Engineer in the Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached.

2. I have reviewed the Joint Intervenors' Motion to Augment or, in the Alternative, To Reopen the Record, dated February 14, 1983, Part E and the appended affidavit of John H. Cooper of January 19, 1984 concerning perceived deficiencies in the design of the Diablo Canyon Residual Heat Removal System.

3. Mr. Cooper's affidavit concerning perceived deficiencies in the design of the Diablo Canyon Residual Heat Removal System is essentially a reiteration of his technical concerns documented in Allegations No. 37 through 45 and 177 with a few new items not previously addressed. My response to his affidavit is divided into the following five groups of technical concerns:

- (a) Effectiveness of natural circulation cooldown and the scope of the natural circulation tests at Diablo Canyon (Issue had not been previously addressed)
- (b) The changes made by PG&E to the Diablo Canyon operating procedure B-2:11 requiring power removal from the RHR isolation valves after these valves were open conflicts with the PG&E commitment stated in the staff SSER No. 7 with regard to achieving cold shutdown from the control room. (Issue had not been previously addressed)
- (c) The RHR hotleg suction line should be designed to safety related requirements. The use of RHR system as a part of ECCS during LBLOCA and SBLOCA. TMI-2 experience with RHR systems. (Portions of these issues had not been previously addressed)
- (d) Effects of inadvertent/spurious closure of RHR hotleg suction isolation valves. RHR system design relative to GDC 34. Spurious closure of the RHR suction isolation valves is a frequent event. (Issues have been previously addressed in Allegation No. 37 through 45 and 177)
- (e) The Staff responses to Allegation 40, contained in SSER 21, is inadequate and further analyses/review are needed. (Coming from *Attachment 1 to this affidavit c/c* Exhibit 17b of Mr. Cooper's affidavit) At the end of this affidavit, I have summarized my response to the allegations 40, 45, 177 and to the concerns in Mr. Cooper's affidavit.

4. Mr. Cooper's Concerns

Mr. Cooper has expressed on Pages 7 and 9 of his affidavit, concerns regarding the effectiveness of natural circulation and the scope of the natural circulation tests at Diablo Canyon Plant.

Relevant Allegation Number

None

Staff Response

Mr. Cooper has expressed concerns regarding the effectiveness of RCS cooling by natural circulation and the adequacy of the natural circulation tests to be conducted during the low power test program. His concerns appear to be based on statements made in draft Reg. Guide 1.139 that indicate natural circulation cooling is "a poor alternative" for core cooling. Draft Regulatory Guide 1.139 has not been issued as a final document and does not represent any final staff position. The Staff's current judgment differs from the statement from the draft Regulatory Guide quoted by Mr. Cooper on page 9 of his affidavit.

A significant amount of information exists to support the efficacy of natural circulation in PWRs. Tests have been run in both the LOFT and Semiscale facilities which demonstrate the ability of analytical models to predict natural circulation. In addition, tests have been run in Westinghouse plants recently licensed (e.g. Sequoyah and North Anna) which have demonstrated this ability to remove decay heat by natural circulation. Operational events have also showed that natural circulation is a viable, effective means of decay heat removal.

With respect to natural circulation tests, the detailed procedures for the natural circulation and boron mixing test for Diablo Canyon plant are currently under the staff review. This test will demonstrate whether or not Diablo Canyon can be brought to the cold shutdown conditions. Even though Diablo Canyon was not reviewed against the BTP RSB 5-1 natu-

ral circulation test requirements, the tests being proposed essentially meet the RSB 5-1 position. The low power natural circulation tests that Mr. Cooper referred to on page 7 of his affidavit are designed for operator training per the requirements of Item I.G.1 of NUREG-0737. The viability of the natural circulation cooldown in PWRs was previously addressed in the NRC staff testimony of W. Jensen regarding Contentions 10 in the Diablo Canyon full power hearing.

Following a loss of offsite power, decay heat removal through secondary system is essential until the primary coolant pressure and temperature reaches the conditions which permit RHR system initiation. As long as the condensate supplies are available to the auxiliary feedwater pumps, the decay heat generated from the reactor core and the sensible heat in the RCS could be continuously removed through steam generators with sufficient natural circulation in RCS.

Staff Conclusion

In summary, it is the Staff's conclusion that cooldown by natural circulation is a viable, effective means of decay heat removal that has been extensively demonstrated. Moreover, Mr. Cooper has not identified any specific problems or issues that can be specifically addressed.

5. Mr. Cooper's Concern

On pages 10 and 11 of Mr. Cooper's affidavit, he discusses the apparent inconsistency between the PG&E commitment addressed in Staff SSER No. 7 and the Diablo Canyon operating procedure B-2:11. SSER No. 7 states that all operator actions needed to perform plant cooldown can be

accomplished from inside the control room. However, operating procedure B-2:11 requires an operator to leave the control room to manipulate the breakers for valves 8701 and 8702.

Relevant Allegation Number

None

Staff Response

At the time the Staff SSER No. 7 was issued in May 1978, there were no procedural requirements to remove power from valves 8701 and 8702 in any mode of plant operation. The applicant intended to always have power available to valves 8701 and 8702. Thus, a plant cooldown could be conducted without any operator actions outside the control room. This design was consistent with BTP RSB 5-1 which had just been implemented by the Staff. As a result of fire protection review PG&E was requested by the Staff to remove power from the RHR suction line isolation valves 8701 and 8702 during power operation. This is discussed in SSER Nos. 8 and 9. This action was intended to ensure that a fire in the vicinity of the RHR isolation valve control circuitry during normal operation would not cause spurious opening of these valves and thus initiate a LOCA outside of containment. Also as described in the Staff response to allegation 45, PG&E was requested to remove power from these isolation valves after being opened. This action was required to prevent spurious closure of the valves for RHR pump protection and to reduce the possibility of a low RCS temperature overpressure event.

Thus there are two manipulations of the 8701 and 8702 motor power supply breakers, both requiring operator actions outside of the control room. The first action, re-instating power to the isolation valves in preparation for valve movement to initiate RHR cooling, was found to be necessary as a result of a fire protection review. The second action, removal of power from the isolation valves after being opened (during shutdowns) was based on RHR pump and low RCS temperature overpressure concerns.

It should also be noted that BTP RSB 5-1 is not considered a requirement, but rather is one acceptable means of meeting the Commission's regulations. Staff experience in implementing BTP RSB 5-1 has shown that in some instances it is necessary to allow, on a case by case basis, limited operator actions outside of the control room to achieve cold shutdown.

Staff Conclusion

The position stated in SSER No. 7 with respect to the ability to conduct a plant cooldown from the control room and the requirement to remove power from the RHR isolation valves during normal plant operation are conflicting requirements.

However, the requirements for power removal from the isolation valves during normal plant operation and shutdown cooling mode are acceptable and the deviation from the BTP RSB 5-1 is acceptable. As discussed in the response to allegation 45, after installation of the low flow alarms, the Staff is requiring that the power remain on the isolation valve during RHR cooling for protection against LOCA's outside of

containment. As discussed in SSER Nos. 8 and 9, power removal from the isolation valves during normal plant operation will continue for fire protection considerations.

The actions that need to be taken by an operator outside of the control room and the time necessary to take them have been examined and have been determined to be acceptable. The applicant has verified that it will take less than five minutes for operators to reach the motor control center one level below the control room and a short distance away to manipulate the breakers. Also the operator will not be exposed to any unacceptable environmental conditions by going to the motor control center.

6. Mr. Cooper's Concern

On page 3, 4 and 5 of Mr. Cooper's affidavit, he states that flow from the RCS hot leg to the RHR system through the single inlet would be required for mitigation of a small break LOCA, as was evident during the TMI-2 accident. Therefore, he asserts that the RHR suction line from the RCS hot leg should be redundant.

Relevant Allegations Allegation Nos. 40 and 177

Staff Response

As the Staff indicated in the response to Allegation No. 40, a large portion of the RHR system is designed to serve a dual purpose and, as was stated in the Staff response to Allegation 40, the ECCS portions of the RHR system at Diablo Canyon meet the single failure criterion. As a part of the ECCS, the RHR pumps take suction initially from the RWST and later,

during the recirculation mode, from the containment emergency sumps. This portion of the RHR system is designed to provide injection or long term recirculation following a large break LOCA. For a small break LOCA, the RHR system does not provide injection since the RCS pressure is normally well above the shut-off head of the RHR pumps. However, as in the case of the large break LOCA, the ECCS portion of the RHR system allows long term recirculation. Contrary to Mr. Cooper's statements, more than enough coolant would be available in the containment sumps for this mode of operation. If the RWST inventory has been reduced by continued injection, the fluid lost through the break will be available in the containment sumps.

Mr. Cooper has cited the TMI-2 accident as an example of a SBLOCA where the RHR system was used. This is incorrect. The RHR system was never relied on for injection, long term recirculation or decay heat removal. (i.e., suction from the RCS hot leg). Decay heat removal was initially accomplished by using the steam generators and the auxiliary feedwater system.

Staff Conclusion

The portion of the RHR system relied on for ECCS function have been reviewed and approved by the Staff, and are in conformance with 10 C.F.R. 50.46 and Appendix K. The RHR hotleg suction line is not a part of ECCS and is not required for mitigation of any size LOCA. The hotleg suction line is used for plant cooldown only.

As was stated in the response to Allegation 40, the Diablo Canyon design with a single RHR suction line meet the position of SRP 5.4.7 and

Branch Technical Position RSB 5-1 for a Class 2 plant. USI A-45 is performing further assessments of the reliability of various decay heat removal system designs.

7. Mr. Cooper's Concerns & Staff Responses

A number of other concerns were expressed in the affidavit that have been addressed by the Staff previously. These are listed below.

- (a) On page 2 of the affidavit, Mr. Cooper discusses the possibility of inadvertent closure of the RHR suction isolation valves. The Staff has addressed this concern in the response to Allegations 45 and 177.
- (b) On page 3 of the affidavit, Mr. Cooper discusses the concern that the single suction line does not meet GDC 34. The Staff addressed this concern in the response to Allegation 40.
- (c) On page 8 of the affidavit, Mr. Cooper states the belief that spurious closure of the RHR suction line is a "recurring common cause" fault that can cause both safety related RHR pumps to fail. The Staff response to Allegation 177 specifically addresses this concern. Also, the Staff has discussed the necessity for a low flow alarm in the response to Allegation 45.

8. Mr. Cooper's Concerns

In Exhibit 17b. Mr. Cooper has the following comments to the Staff response on Allegation No. 40.

- (a) The RHR system should be reanalyzed in the light of the TMI accident.

- (b) Two suction lines from the RCS to the RHR are provided in CE, B&W, and the newer Westinghouse designs.
- (c) In the proposed Regulatory Guide 1.139, the RHR system is required to be redundant, and withstand any type of an accident, not just a large break LOCA.
- (d) Accessibility of the RHR isolation valves inside containment during radiological conditions.

Relevant Allegation Number

Allegation No. 40 and 177

Staff Response

My responses to Mr. Cooper's comments are as follows:

- (a) The RHR system was not used during TMI-2 post accident operation. The lessons learned from TMI-2 on the RHR system are irrelevant to the subject issue (single failure concern).
- (b) The early designs of Westinghouse, CE, and B&W plants all have only one RHR suction line from RCS hotleg. The current designs are equipped with redundant suction lines. The NRC Unresolved Safety Issue A-45 is assessing the adequacy of the RHR design with respect to the single suction line and the pressure interlock features on the suction isolation valves.
- (c) Proposed Regulatory Guide 1.139 has not been finalized or issued by the Staff.
- (d) PG&E has informed the Staff that the radiological conditions in the vicinity of the RHR valves are acceptable for the operator to enter

the area during normal plant shutdown conditions. During accident conditions, decay heat should be removed via steam generators and the auxiliary feedwater systems or after a LOCA, by the RHR system in the recirculation mode.

Conclusion

As discussed in item above, the RHR system was not used, nor was it necessary during the TMI-2 accident. The single suction line design, which is common in most operating reactors, has been reviewed and approved at Diablo Canyon, and is acceptable. The generic implications of RHR system single suction line are part of the ongoing Unresolved Safety Issue A-45.

The Staff's current judgment differs from and does not endorse the statements in the draft Regulatory Guide 1.139, as described in the item above. The Regulatory Guide 1.139 is still in draft form. The positions stated in the draft do not reflect current regulatory requirements or positions.

Overall Summary

Mr. Cooper has raised a number of questions and concerns centered around the adequacy of the Diablo Canyon RHR system single suction line. The Staff has addressed each of these concerns in its responses to Allegations 40, 45 and 177, as well as in the discussions provided above.

In summary, the Staff believes the RHR single suction line design is acceptable, and USI A-45 will assess the overall reliability of decay heat removal systems.

The Diablo Canyon design has the capability to remove decay heat without the RHR system by relying on the steam generators and the auxiliary feedwater system. There are adequate water supplies available to the auxiliary feedwater system, and there is the capability to utilize backup water supplies should the safety related condensate storage tank (CST) supply be depleted.

The Staff believes that natural circulation cooling of the RCS is viable, and there is sufficient operational experience, experimental data and analytical calculations to confirm the validity of the process.

Mr. Cooper is incorrect in his assertion that the RHR system single suction line must be available for mitigation of certain LOCAs and that this path was used during the TMI-2 accident. The Diablo Canyon design places no reliance on this flow path for any LOCA scenario, nor was this path used in the TMI-2 accident.

Furthermore, the Staff believes that installation of the RHR low flow alarm will provide positive indication to the operator should either RHR suction line isolation valve inadvertently close while the RHR pump(s) are operating. The installation of this alarm will be completed prior to power operations.

I hereby certify that the answers are true and correct to the best
of my knowledge.

Chu-yu Liang
Chu-yu Liang

Subscribed and sworn to before me
this 15th day of March 1984

Michael J. Hadden
Notary Public

My commission expires: July 1, 1986

Professional Qualifications

Chu-yu Liang
Reactor Systems Branch
Division of Systems Integration
U.S. Nuclear Regulatory Commission

I am employed as a Senior Nuclear Engineer, Reactor Systems Branch, Division of Systems Integration, U.S. Nuclear Regulatory Commission, Washington, D.C. The Reactor Systems Branch is responsible for reviewing reactor license applications and evaluating the design of reactor systems, including the residual heat removal and emergency core cooling systems, of the nuclear power plant with respect to nuclear safety. As part of my duties, I have been responsible for reviewing the operating license applications of several PWR facilities with respect to reactor systems.

From 1965 to 1967, I was employed by Lockwood, Andrews and Newman, Inc. (Houston, Texas), where I worked on the design of mechanical systems for public buildings including heating, ventilation and air conditioning systems, central plant and emergency power systems.

From 1967 to 1969, I was employed as a mechanical engineer by Avondale Shipyards, Inc. (New Orleans, Louisiana), where I worked on the design of marine steam power plants for tankers, destroyers, and cargo ships.

From 1969 to 1974, I was employed as a Senior Engineer in the Department of Systems Engineering, PWR Systems Division, Westinghouse Electric Corporation (Monroeville, Pennsylvania), where I worked on the design and review of nuclear power plant auxiliary and power conversion systems. I served as a lead engineer for 16 Westinghouse PWR plants,

providing balance of plant design criteria and NSSS interface requirements and assisting plant designers (e.g., Architect-Engineers) in the areas of auxiliary and power conversion system design.

From 1974 to the present, I was employed by the AEC, in the Auxiliary and Power Conversion Systems Branch, Division of Technical Review; following the reorganization of the AEC, I served as a systems engineer in the Auxiliary Systems Branch, Division of Systems Safety, U.S. Nuclear Regulatory Commission. In 1980, I commenced employment with the Reactor Systems Branch, Division of Systems Integration.

I attended the Cheng-Kung University, Taiwan, and received a B.S. Degree in Mechanical Engineering in 1960. I received a Master of Science Degree in Mechanical Engineering (majoring in steam power plant design) from the Oklahoma State University in 1965. I have also attended the Graduate School of Engineering at Catholic University, Washington, D.C., where I took a course in Nuclear Engineering.

I am a member of the American Society of Mechanical Engineers.

SER REBUTTAL 1-10-84

On January 7, 1984, I received a copy of NUREG 0675, Supplement No. 21 - the Safety Evaluation Report related to the operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2. This document contains the official NRC responses to the "allegations" (NRC's terminology) which I made to Mr. Eugene Powers of the NRC Office of Inspection and Enforcement on August 20, 1981, almost two and one-half years ago.

One would think, given the amount of time available to the NRC to evaluate my concerns, that their engineers would have done a thorough job of evaluating them. From reading the appropriate portions of this supplement (Allegations #37-45), however, it becomes apparent that this document has been hastily prepared during the last few weeks in an attempt to mollify the growing public concern over the large number of outstanding problems at Diablo Canyon. The number of typographical errors contained in this document gives some indication of the amount of time spent in it's preparation. But more serious, to me, is the lack of responsiveness to my original questions, and the lack of justifications supplied by the NRC for what I consider to be inadequate answers. NRC personnel, in their "responses" to my concerns, continue to ignore the basic facts of my case, the operating experiences at Diablo Canyon and at other nuclear power plants, and they continue to insist that, Code of Federal Regulations notwithstanding, repeated malfunctions in a safety-related system are not considered to be a "significant safety concern" unless the system malfunctions when called upon to actually perform it's safety function during an accident. I am convinced - and I think most thinking individuals would agree - that the time to correct problems with a safety-related system is before it is actually needed to prevent an accident or safely shut down the plant.

NRC representatives have told me that if the malfunctions about which I am concerned had happened when there was fuel in the reactor, or when the system was called upon to function, then they would be considered a significant safety concern. Since, however, the system failed (twice) before fuel was loaded, there was no threat to the health and safety to the public, therefore no safety problem. This philosophy is legally, morally, and logically bankrupt, and is akin to saying that a high-speed automobile bearing down on a pedestrian is no threat to his health and safety until it actually hits him.

The following paragraphs contain my analyses of, rebuttals to, and comments on the NRC "responses" (or actually, lack of responses) to my "allegations".

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ALLEGATION No. 37 "The solid state protection system (SSPS) relays that initiate closure of RHR letdown isolation valves 8701 and 8702 perform no safety function, reduce the reliability of the RHR system, and cause a potential for RHR pump damage. Therefore, these relays should be removed."

The NRC response to this concern is very disturbing, since it displays a total lack of understanding of how this system functions. To me, this is especially disturbing since this response is from the NRC engineering staff in Washington, DC - the supposed "experts" on nuclear power plants.

In the first place, the SSPS does not initiate the closure of valves 8701 and 8702. The automatic closure of these valves is initiated by the changing of state of a "comparator" module located in the Protection and Control Racks in the cable spreading room (directly below the control room). From there, the circuitry passes through four input relays in the SSPS (located in a separate room, adjacent to the control room) and then back down to the cable spreading room to the Auxiliary Safeguards Racks, and then to the motor control centers for the valves. The SSPS in no way amplifies or changes the signals, or performs any logic function with them - the signals merely pass through the relays in the SSPS. Why PG&E continues to insist that "The solid state protection system completes the logic function and generates a larger output signal (amps.) which in turn actuates relays in the auxiliary logic cabinet" and the NRC continue to insist that "This automatic isolation function [is] performed by the Westinghouse designed SSPS" is truly amazing. Certainly both of these organizations have access to the circuit diagrams for this system and the expertise to understand them. One might almost conclude that this is a deliberate attempt to make these circuits appear to be a part of the engineered safety features of the SSPS when they really aren't.

Secondly, both PG&E and the NRC continue to insist that the RHR system has a temperature/pressure interlock system to automatically close valves 8701 and 8702 if the temperature or pressure in the reactor coolant system exceed predetermined values, thus preventing an "inter-system LOCA". Both PG&E and the NRC are aware that the Diablo Canyon Technical Specifications require that the power be removed from the actuators for these valves during the period when the automatic closing action would be desired. It is a mystery to me how both organizations continue to make this claim when they know that the power is removed from the valve actuators, preventing them from automatically closing; unless, again, there is a deliberate attempt to mislead the public.

The NRC asserts that "diverse indications and alarms are provided in the control room (including a RHR system low flow alarm to be installed during the first refueling outage) to allow the operator(s) to assess RHR system status and to alert them to potential system degradation." George Orwell would be proud of the author of this sentence; it is such a fine example of "newspeak". In the same sentence, the presence of an alarm is claimed, and the schedule for installing it is given. Perhaps the NRC would like to explain how it was possible, with all these

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"diverse indications and alarms" for an RHR pump to be run without suction, unnoticed by the operators, for one hour, until it was damaged, as happened at Diablo Canyon last October. (The pump had to be replaced.)

Lastly, the NRC states that "this allegation does not involve considerations that question plant readiness for power ascension testing or full power operation", and yet goes on to say, several pages later, that "during the first cycle of operation, plants operate more frequently on the RHR system, testing and training requirements for a new plant. Thus, the period of vulnerability to a spurious RHR suction MDV closure may be greater than in subsequent cycles." Apparently, the NRC sees no contradiction in these two statements.

ALLEGATION No. 38 PG&E is ignoring evidence that the spurious closure of a motor operated valve is not "impossible".

The NRC's position, here, was very well-timed. They state: "The staff has examined in depth the licensee's actions in response to an event involving the spurious initiation of RHR motor operated valve closure as well as the concerns expressed by the allegor regarding the potential for such event (sic), and concluded that timely evaluation and corrective measures were taken to preclude repetition of such conditions." In the few weeks since this statement was made, another "spurious initiation of RHR suction valve closure" at Diablo Canyon caused damage to an RHR pump. This is twice now that the NRC has proclaimed that the problems with this system have been "resolved" after making only paper changes. How many more times will the Diablo Canyon RHR pumps have to suffer damage before the people in charge out there realize that the problem is inherent in the circuitry, not the procedures? As of today, PG&E and the NRC continue to espouse the same philosophy which has in the past led to pump damage and loss of decay heat removal capability at Diablo Canyon and at many other plants around the country.

Over three years ago, I gave PG&E copies of 16 Licensee Event Reports documenting cases of "spurious initiation of RHR suction valve closure" in various plants around the country. I'm sure that many more cases must have occurred since that time besides the latest one at Diablo. But PG&E still continues to contend that "A failure, such as the spurious closure of a motor operated valve...has not been considered credible." (FSAR, page 3.1-3), and that "Westinghouse does not consider spurious operation of electrically controlled valves as a credible single active failure" (FSAR, page 15.4-8), and that "The probability of any spurious valve closure is therefore 2.54×10^{-8} to the minus 8th power per valve-hour." (FSAR, page 6.3-34a). In the face of the overwhelming evidence that spurious valve closures happen quite regularly, I can only interpret the failure of PG&E and the NRC to recognize this evidence (and act upon it) as yet another attempt to mislead the public as to the safety of the RHR system.

The NRC states that "It does appear that the licensee is giving proper attention to the spurious closure of the valves in question". I would like to point out that mere attention is only

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the first step toward correcting the deficiencies in the RHR system. Action is necessary to actually correct the problem and prevent it from happening again. How many more times will an RHR pump have to be damaged at Diablo Canyon before the necessary action - wiring the interlock circuitry directly from the "hagan racks" to the "Auxiliary Safeguard Racks", without it passing through the SSPS, and adding the low RHR flow alarm - will be taken?

ALLEGATION No. 39 "There is no control room annunciation provided to alert the operator(s) when the RHR letdown line has been isolated during Modes 4, 5, and 6 (hot shutdown, cold shutdown, and refueling respectively).

PG&E was instructed, on April 2, 1981, to install a low RHR flow alarm in the control room, but was allowed to wait until after the first refueling outage to do so. Apparently PG&E intends to wait until the last permissible minute to install this alarm, even though its presence would have prevented the damage to the RHR pump which occurred two months ago. This attitude does not boost my confidence in PG&E's commitment to the safe, reliable operation of Diablo Canyon.

The NRC intends to adhere to their original schedule for the installation of this alarm, even in the face of this second incident, stating: "The staff has concluded that the existing control room indications and procedures are sufficient to assure adequate decay heat removal in the interim."

ALLEGATION No. 40 "The question raised was with regard to whether or not the single RHR pump suction line from the RCS hot leg meets safety related standards. The newer PWRs are designed with redundant RHR pump suction lines from the RCS hot legs."

Again, the NRC missed the boat on this one. My contention was that this system should be reanalyzed in the light of the TMI accident. PG&E claims that the single RHR suction line is not safety related and is only used during the normal cooldown of the plant. I disagree. At TMI, their RHR system was used to mitigate the consequences of the now famous accident there. I propose that this portion of the Diablo Canyon RHR system is inadequate, since a single failure in this line would prevent decay heat removal via this system. I offer these facts as evidence:

1. The proven unreliability of the suction valves in this line, both at Diablo Canyon and at other plants.
2. That two, safety related suction lines from the RCS to the RHR are provided in CE, B&W, and the newer Westinghouse designs. Why would these extra lines be provided if they weren't needed?
3. In Regulatory Guide 1.139, the NRC staff states that the Residual Heat Removal System is required to be redundant, and

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withstand any type of an accident, not just a large break LOCA.

Apparently, the NRC is aware of the deficiencies in this and perhaps other areas in the design of Diablo Canyon, since they consider it a "Class 2 Plant". They go on to state that "A single RHR suction line from the RCS hot leg is considered acceptable for a class 2 plant as long as a single failure could be corrected by manual actions inside or outside of containment, or the plant could be returned to hot standby until manual actions (or repairs) are accomplished." I would like to pose a question at this point. How, under radiological conditions which would prevent human entry into the containment (such as were found in the containment at TMI), could valves 8701 or 8701 be opened manually or repaired? Both of these valves are located inside the containment structures at Diablo Canyon.

ALLEGATION No. 41 The power source of certain relays is not shown on certain drawings and this caused an operational problem, the failure (closure of RHR isolation valves)"

Although the contacts of the SSPS input relays are shown on the electrical schematic of this system, the power source for the relay coils is not shown on any approved plant drawing. The NRC continues to state that the September 1981 incident was due to a "lack of pre-planning" implying that if the technician responsible for the incident (and his foreman) had just done their job properly, the spurious closure would not have occurred. It seems to me, that if the information is not on any drawings, then no amount of pre-planning will help, and errors are bound to be made. The NRC takes great pride that a technician-drawn "composite drawing" of this system has been put together. I put together a similar drawing back in May of 1981, and gave it to the instrument foreman, but that obviously didn't prevent the incident 3 months later. Hand drawn, or "bootleg" drawings can get lost, or not be distributed to the persons who need the information. That is the whole purpose of the Drawing Control System - to ensure that accurate, up to date information is readily available and accessible to those who need it. Not only is this idea good common sense, but it is part of the Code of Federal Regulations:

10CFR50 Appendix B, III Design Control

Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in paragraph 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions."

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ALLEGATION No. 42 Licensee management was unresponsive to recommendations to prevent spurious closure of the isolation valves on the residual heat removal (RHR) system. Closure of the valves disables operation of the RHR system for decay heat removal.

The chronology of events in this case speaks for itself. At no time did PG&E take any effective action to resolve any of my concerns without the intervention of the NRC. Most of the the original problems which I brought first to PG&E's, then the NRC's attention remain uncorrected even today, after three years of unsuccessful attempts on my part.

In the NRC assessment of the safety significance of this problem, an attempt is being made to give the impression that PG&E, Westinghouse, and the NRC have all been actively working for the past three years solve these problems. I find this very hard to believe, since no evidence of any kind has been produced to support this position. As far as I'm concerned, the last consideration that this problem received was in November of 1981, when the Diablo Canyon Onsite Review Group decided to take no action to correct the problem.

ALLEGATION No. 43 The loss of the residual heat removal (RHR) system on 9/29/81 due to unplanned closure of the RHR isolation valves was an event which should have been reported to the NRC in accordance with 10CFR50.72. The licensee's failure to make such a report was in violation of NRC regulations.

10CFR50.72 Notification of significant events

(a) Each licensee of a nuclear power reactor licensed under paragraph 30.21 or 50.22 of this part shall notify the NRC Operations Center as soon as possible and in all cases within one hour by telephone of the occurrence of any of the following significant events and shall identify that event as being reported pursuant to this section:

"(6) Personnel error or procedural inadequacy which, during normal operations, anticipated operational occurrences, or accident conditions, prevents or could prevent, by itself, the fulfillment of the safety function of those structures, systems, and components important to safety that are needed to... (i) remove residual heat following reactor shutdown..." (my emphasis)

The NRC claims that "The loss of residual heat removal capacity during a time when significant fission product decay heat is present in the core would have safety significance. In this particular instance, fuel had not been loaded into the Diablo Canyon Unit 1. Therefore, no fission product decay heat was present and loss of RHR capability had no actual safety significance." Again, the NRC doesn't perceive any safety problem until the speeding automobile actually hits the pedestrian.

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ALLEGATION No. 44 The licensee failed to properly process a Nuclear Plant Problem Report.

In my original complaint I stated that the Nuclear Plant Problem Report documenting the first loss of RHR suction at Diablo Canyon was "signed off as complete without any plant management review...classified as 'non-reportable' and without any follow-up action such as an RHR pump inspection or investigation into the cause of the event." The NRC admits that the above concerns are true (although two months after the event, another problem report was initiated to perform a pump test), they just believe that this is an acceptable way to run a power plant. In my discussions with Mr. Jess Cruse of the NRC, who interviewed the "principles" in the handling of this problem report, he stated that "no-one denies that it could have happened just the way you said it [did], and I sort of concluded most likely it did happen [that way]". What action did Mr. Cruse take? None. Mr. Cruse also stated that this was not 'reportable' because there was no fuel in the reactor at the time. Again, it would seem to me that the NRC would wish to know of problems before there was fuel in the core, but this is apparently not so. As for the analysis of the problem to prevent reoccurrence, this has not been done by PG&E even to this day, as evidenced by the recent (Nov. 83) replay of the September 1981 incident. PG&E and the NRC both claim that "strict procedural controls" are adequate to prevent reoccurrence, although this method has been proved inadequate twice before.

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ALLEGATION No. 45 Section 5.5 of the Diablo Canyon FSAR describes the autoclosure interlock for the RHR suction line isolation valves (8701 and 8702). Section 3.4.9.3.a of the Diablo Canyon Technical Specifications requires power to be removed from these isolation valve operation during modes 4 (Hot shutdown when RCS cold leg temperature is less than 323 degrees F), 5 (cold shutdown), and 6 (refueling). This requirement defeats the function of autoclosure interlock for the valves.

In their lengthy analysis of this simple allegation, the NRC admits that removing the power from these valve operators defeats the autoclosure interlock to the RHR suction valves as described in the FSAR. I contend that either the FSAR should be corrected so it accurately describes the RHR system at Diablo Canyon, or the RHR system should be operated in conformance with the FSAR. The Code of Federal Regulations is clear in both cases: The NRC must be notified if the plant design does not conform to the criteria and bases in the FSAR, and the FSAR must be kept up to date.

10CFR50.55 Conditions of construction permits

(e)(1) If the permit is for construction of a nuclear power plant, the holder of the permit shall notify the Commission

of each deficiency found in design and construction, which, were it to have remained uncorrected, could have affected adversely the safety of operations of the nuclear power plant at Any time throughout the expected lifetime of the plant, and which represents:

(ii) A significant deficiency in final design as approved and released for construction such that the design does not conform to the criteria and bases stated in the safety analysis report or construction permit

10CFR50.71 Maintenance of records, making of reports

(e) each person licensed to operate a nuclear power reactor pursuant to the provisions of paragraph 50.21 or 50.22 of this part shall update periodically, as provided in paragraphs (e)(3) and (4) if this section, the final safety analysis report (FSAR) originally submitted as part of the application for the operating license, to assure that the information included in the FSAR contains the latest material developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement since the submission of the original FSAR or, as appropriate, the last updated FSAR. The updated FSAR shall be revised to include the effects of: all changes made in the facility or procedures as described in the FSAR; all safety evaluations performed by the licensee either in support of conclusions that changes did not involve an unreviewed safety question; and all analyses of new safety issues performed by or on behalf of the licensee at Commission request. The updated information shall be appropriately located within the FSAR.

(3)(1) A revision of the original FSAR containing those original pages that are still applicable plus anew replacement pages shall be filed within 24 months of either July 22, 1980, or the date of issuance of the operating license, whichever is later, and shall bring the FSAR up to date as of a maximum of 6 months prior to the date of filing the revision.

The NRC, in it's analysis, curiously avoids any mention of the above two regulations, but goes on to say that operating the plant with the power removed from these actuators is a violation of their Branch Technical Position RSB 5-1, Position B.1.C. They also state that "There have been many occasions of spurious RHR suction valve closures on [sic] operating plants. This has resulted in not only a loss of decay heat removal, but also an overpressure event due to the loss of the letdown flowpath." They continue that "During the first cycle of operation, plants operate more frequently on the RHR system as a result of maintenance, testing and training requirements for a new plant. Thus, the period of vulnerability to a spurious RHR suction MOV

closure may be greater than in subsequent cycles. They basically agree with everything I've been saying for all these years - and their "response" is to do nothing to correct the situation until the first refueling outage. I never cease to be amazed at the workings of the bureaucratic mind.

Task: Allegation 177

ATN No. None

BN No.: None

Characterization

The allegation relates to the RHR pump common suction line valve control and a potential damage to RHR pumps due to loss of suction as a result of a single failure.

Related Allegations: 37, 39, 40, 45 (previously discussed in SSER 21)

Implied Significance to Plant Design, Construction or Operation

The RHR suction line from the RCS hot leg in the Diablo Canyon design contains two isolation valves (8701 and 8702) in series that are normally closed during power operation and hot standby condition (Modes 1, 2 and 3). The RHR suction line from the RCS hot leg is only used during Mode 4 (hot shut-down with RCS cold leg temperature less than 323 °F), Mode 5 (cold shutdown) and Mode 6 (refueling). A postulated inadvertent closure of either isolation valve (8701 or 8702) in the RHR suction line during plant shutdown could cause potential damage to both RHR pumps.

Assessment of Safety Significance

This allegation overlaps concerns previously expressed in Allegations 40 and 45 which have been addressed by the staff in Diablo Canyon SSER No. 21.

This concern also has been discussed by the staff at an ACRS meeting on February 10, 1984.

The potential damage of both RHR pumps due to loss of suction as a result of a single failure is prevented by the following provisions:

1. In response to the staff requirement in SSER 21 regarding Allegation 45, PG&E has committed, in a letter dated February 15, 1984, to install the RHR low flow alarm prior to entry into power operation (i.e. Mode 1 with associated decay heat generation). The low flow alarm will be set so that sufficient time would be available to alert the operators to trip the RHR pumps before pump damage occurs.
2. The current Technical Specifications and operating procedures for Diablo Canyon Unit 1 preclude the inadvertent closure of either of the two RHR pump suction line isolation valves (8701 and 8702) by maintaining the valves in an open position with power removed for the valve operators during Modes 4, 5 and 6.

The applicant stated at the ACRS meeting on February 10, 1984 that RHR pump damage could occur in 10 to 15 minutes following loss of suction flow.

Operating experience from the Calvert Cliffs Nuclear Power Plant showed that the RHR pump seals were damaged approximately 15 minutes after loss of suction flow. The failure of both RHR pumps is an event beyond the design basis and its occurrence is highly unlikely based on the plant specific design and administrative controls discussed above. However, if failure of both RHR pumps should occur during plant shutdown, the following steps could be taken to maintain a safe shutdown condition:

1. If both RHR pumps failed during the period when the decay heat level is still relatively high, then the plant conditions would permit decay heat to be removed by the steam generator(s). Condensate supplied from the condensate storage tank, raw water reservoir, and the auxiliary salt water system (unlimited supply) via temporary connections could provide a long term source of auxiliary feedwater for decay heat removal.
2. If the steam generator(s) were not available, and the decay heat is relatively low, one RHR pump is generally used to remove decay heat with one pump in standby, in accordance with the requirements of Technical Specifications 3.9.8.2. In case the operating RHR pump is damaged due to closure of a suction valve, the standby RHR pump could be used to continue the decay heat removal function after the closed suction isolation valve(s) is manually opened by an operator. Analyses indicate that if all decay

heat removal capability were lost at the time of reactor trip, at least 2 hours would be available for the operators to restore decay heat removal capability before core uncover. If decay heat removal capability were lost while on RHR cooling, considerably more time than 2 hours would be available for operator action to correct the situation.

3. If both RHR pumps were damaged while the steam generators were open for maintenance (or during any other period in which all steam generators were unavailable), the charging pumps or safety injection pumps could be used to inject water into the RCS for core cooling. If the manways on the steam generator primary side were open for maintenance, water would flow out the manways and onto the floor of the containment. The containment spray system and the fan coolers, which are independent from the RHR system, could be used to remove decay heat inside containment to the ultimate heat sink via the component cooling water or the essential service water system.
4. Diablo Canyon Operating Procedure No. EOP-17 addresses the emergency procedure under the condition that both RHR pumps are damaged during plant shutdown.

In summary, the staff recognizes that closure of either of the two isolation valves in series in the RHR hot leg suction line would prevent the RHR system from performing its decay heat removal function and could result in damage to the RHR pumps if not corrected. Our evaluation has concluded that:

- a. Although the staff did not specifically evaluate the Diablo Canyon RHR system against the criteria of BTP RSB 5-1 at the time the system was reviewed, the staff concludes that the system meets the intent of BTP RSB 5-1 for Class 2 plant implementation. The only deviation we have identified is the lack of a qualified auxiliary feedwater supply in excess of 8 hours. However, there are other diverse auxiliary feedwater sources available, which, while not designed to safety grade standards, nonetheless provide a high degree of assurance that an ample auxiliary feedwater supply will be available.
- b) Technical Specifications and administrative procedures are in place at the plant to assure that the two series isolation valves in the RHR suction line are locked open with power sources removed from the valve operators. Moreover, a RHR low flow alarm will be installed and made operational prior to power operation to ensure that the operators will be alerted to any low flow condition that would occur in the RHR suction line, such as could occur from a closed isolation valve. Given spurious isolation valve closure as an initiating event, the failure of the operators to follow administrative procedures and technical specifications, combined with a failure of the low flow alarm or the operators to take corrective action in the presence of a low flow alarm must be postulated in order for RHR pump damage to result.

The staff considers that the need to postulate two independent failures to lose the RHR capability meets the intent of the single failure criteria.

The above capability combined with the additional capabilities to remove the decay heat even if the RHR system were lost, lead the staff to conclude that the RHR design of the Diablo Canyon Plant does not pose undue risk to the health and safety of the public.

The staff is currently conducting a generic re-evaluation of the requirements for shutdown decay heat removal systems. This work is being performed under Unresolved Safety Issue (TAP A-45). The effort includes a reassessment of the adequacy of the single RHR suction line from the hot leg and the interlocks on the suction line isolation valves.

Staff Position

Based on the staff evaluation and assessment of the safety significance as discussed above, the staff finds that this allegation does not involve considerations not previously considered for plant readiness for low power or full power operation.

Action Required

No specific action regarding Diablo Canyon is required. The staff is conducting a generic reevaluation as discussed above.