10-20-83
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Mr. T.W. Bishop, Resident
Reactor Projects and Engineering Programs
Region V
1450 Maria Lane
Walnut Creek, CA 94596

Dear Mr. Bishop:

Thank you for your letter responding to my 8-20-82 allegations concerning the RHR (Residual Heat Removal) system at Diablo Canyon. Your office seems to have spent quite a bit of time on this investigation, and I appreciate that fact. I am concerned, however, that much of the information which I gave to Mr. Powers during the interview in August of 1982 was apparently not passed on to the Site Representative at Diablo Canyon. The inspector who "paraphrased" my allegations for Inspection Report No. 50-275/82-42 seems to have "missed the point" on several of them.

I would like to take this opportunity to restate my position on the problems at Diablo Canyon as I previously described them to Mr. Powers. I have attached a copy of "Allegations Regarding the Diablo Canyon Residual Heat Removal System" beginning on page 5 of Inspection Report No. 50-275/82-42, and will comment on these allegations paragraph by paragraph:

(a) I did not claim that there were no control and interlock circuit drawings for valves 8701 and 8702 as your inspection report stated. In fact, I provided Mr. Powers with excerpts from logic diagram 458840, and electrical schematics 437592 and 458846 explaining how this circuit functioned. I pointed out that it was not clearly shown on any of them that removing the power from the SSPS(Solid State Protection System) output relays would cause the RHR suction valves to fail closed. I said that the power source for the SSPS relays in this circuit should be shown on electrical Schematic 437592. The omission of this information from the electrical schematics at Diablo Canyon led to personnel error causing the inadvertent closure of valve 8701 and the isolation of the RHR pumps suction with a pump running (see NPPR DC1-81-OP-P1057 dated 9-29-81).

(b) I did not, at any time during the interview with Mr. Powers, mention the <u>physical</u> routing of the RHR control circuitry. I stated that neither the Senior Control Operator or the I&C Foreman were aware that the RHR control circuitry was routed through the SSPS and that removing the power from the SSPS output relays would cause valves 8701 and 8702 to fail closed. This fact was demonstrated on September 29th as mentioned in the previous paragraph.

(c) This section contains several errors which I have listed below:

1) The SSPS does not amplify, or in any way change the signal from the Westighouse "hagan" racks to the auxiliary relay cabinets. I explained this clearly to Mr. Powers, and gave him a hand-drawn schematic depicting the complete system function for your office to review. The person within PG&E who told you that "the solid state protection system completes the logic function and generates a larger output signal(amps.)..." had best review his basic electricity.

2) That this is a standard Westinghouse design is true, which explains why the Westinghouse reactors are prone to RHR suction valves going closed inadvertantly.

3) That the PG&E management is "unaware of any problems with this arrangement" is curious. On 1-16-82, I submitted Design Change Request No. DC0-GE-2518-Rev.1 to PG&E Engineering. This DCR requested that the RHR system at Diablo Canyon be modified to prevent reoccurrance of the incident of 9-29-81. Attached to it were copies of 16 Licensee Event Reports describing similar losses of Residual Heat Removal capability at other power plants in the Country. This DCR was approved by D.A.Rockwell and R.D.Etzler, both of whom were present during the NRC investigation into these allegations. Copies of these LER's were also provided to Mr. Powers.

(4) The major point which I made to Mr. Powers was that the SSPS relays in this system perform no function whatsoever, reduce the reliability of the RHR system, cause a potential for damage to the RHR pumps, and should be removed, regardless of the fact that this is a "Standard Westinghouse Design".

(d) I pointed out to Mr. Powers that several attempts had been made to "get rid of this problem", and gave him a copy of design change request No. DC0-GE-2518-Rev.1 which I described in paragraph (3) above. I also gave him a copy of Plant Design Comment No. 559 which concerned valves 8701 and 8702. Apparently, your Site Representative confused a "drawing change request" for a "design change request".

(e) I showed Mr. Powers that one portion of the Diablo Canyon FSAR claimed that valves 8701 and 8702 would close automatically on an overpressure/overtemperature condition, while another portion claimed that the power would be removed from these valves during operation. Obviously, a valve can't automatically close when the power is removed from its motor operator. As to the NRC finding that this contradiction in the FSAR presents no "noncompliance with regulatory requirements", may I call your attention to 10 CFR 50.71, paragraph (e), which requires that the FSAR be updated periodically (no less than annually) and shall "reflect all changes up to a maximum of 6 months prior to the date of filing."

(f) PG&E maintains that the spurious closure of a motor-operated valve is essentially impossible. As previously mentioned, copies of the following Licensee Event Reports, all instances of spurious RHR suction valve closures, were given both to Mr. Powers and to PG&E Engineering: LER 369-81072, McGuire-1; 338-79145, North Anna-1; 348-79036, Farley-1; 344-78010, Trojan; 348-80077, Farley-1; 316-80060, Davis-Besse-1; 344-76000, Trojan; 346-80058-1, Davis Besse-1; 316-80013, Davis Besse-1; 339-80001, North Anna-2; 346-77000, Davis Besse-1; 302-80015-1, Crystal River-3; 369-81129, McGuire-1; 317-74000, Calvert Cliffs-1; 348-80080, Farley-1; 318-79038, Calvert Cliffs. Why PG&E continues to ignore this evidence is beyond me.

(g) Yes, an RHR pump motor trip is annunciated in the control room. Unfortunately, the pump motor only trips after the pump has been damaged by overheating due to lack of flow; Yes, the monitor light boxes show RHR suction valve position, but only during accident conditions, not during normal operation. When valve 8701 went closed spuriously on 9-29-81, the Control Room Operator would have remained unaware of the fact until RHR pump failure had not a consciencious painter who was working near the pump called the control room due to the loud banging noises the pump was making. I stand on my original allegation: During normal operation there is no control room annunciation that an RHR suction valve is in the closed position, and there should be one to prevent damage to the RHR pumps.

(h)(j) As I explained to Mr. Powers, Nuclear Plant Problem Report No. DC1-81-OP-P1057 was initiated on 9-29-81, but signed off as complete <u>without any</u> plant management review. When I became aware of this, I contacted Juanito Diamonon, the head of the QC department at the time. NPPR DC1-81-OP-P1057 was resurrected from the "closed" files and signed off by Jim Sexton, but classified as "non-reportable" and without any follow-up action such as an RHR pump inspection or investigation into the cause of the incident. I alleged that both the loss of Residual Heat Removal Capability and the failure to report it were reportable; The former under 10 CFR 50.72 "Notification of significant events", which states that: "Personnel error or procedural inadequacy which, during normal operations, anticipated operations, occurences, or accident conditions, prevents or could prevent, by itself, the fulfillment of the safety funcion of those structures, systems, and components important to safety that are needed to...(ii)remove residual heat following reactor shutdown..." must be reported to the NRC.

(i) I am aware that the Diablo Canyon FSAR claims that the RHR pump suction from the RCS(Reactor Coolant System) Hot Legs is not safety related, but my question is <u>why</u>! This system is certainly necessary to mitigate the consequences of an accident of the small break LOCA type, so why is it not safety related? In the newer Westinghouse and Combustion Engineering designs this system is considered safety related and is totally redundant, so why not at Diablo Canyon?

(k) That the NRC Site Representative considers it "not of safety significance" that a problem report has been open, unresolved, and unreviewed for 3 years is puzzling to me.

(l) PG&E claims no awareness of the incorrect alarm listing in Volume 16 of the Plant Manual. That is really funny, considering that I sent memorandums on this subject to both Mr. J.M. Gisclon, the Power Plant Engineer, and the second to Mr. R.C. Thornberry, the Plant Manager! I gave Mr. Powers copies of these two memorandums during the interview.

Although your office has gone through the motions of an investigation into these problems, it seems to me that the questions which were asked of PG&E were trivial or incorrect representations of those concerns which I conveyed to Mr. Powers. In addition, it appears that PG&E's answers to even those were accepted without question or follow-up. Furthermore, in the instances where I provided Mr. Powers with documents proving my allegations, the documents were not made available to your Site Representative or were ignored.

I can only hope this and the 14 month delay in answering my concerns can be explained merely as a lack of communication between the Office of Inspection and Enforcement, your office, and the Site Representative at Diablo Canyon.

Yours Truly,

John H. Cooper

completed the as-built audit and has generated twenty-nine Open Item Reports (OIR). So far, the licensee has dispositioned eighteen of the OIR's. The inspectors will complete the review in this area when the remainder of the OIR's are dispositioned (82-42-02).

No items of noncompliance or deviations were identified.

8. Open Items Followup

Plant Administrative procedures C451 and D756 have been prepared to assure reinstatement of Environmental Qualification conditions after maintenance or surveillance testing. This closes open items 80-16-01 and TI-15-41.

9. Allegations Regarding the Diablo Canyon Residual Heat Removal System

On December 2, 1982 the inspector met with licensee representatives to discuss allegations regarding the Diablo Canyon residual heat removal (RHR) system. These allegations had also previously been examined at the jobsite and documented in Region V inspection reports 50-275/82-26 and 50-323/82-13. The following paragraphs paraphrase the allegations, summarize the inspection, and state the findings of the inspector.

(a) Allegedly there were no control and interlock circuit drawings for motor operated valves 8701 and 8702 (RHR hot leg suction isolation valves). The inspector examined PG&E drawings 437592 "Residual Heat Removal Flow Control Valves", and 103058 "Circuit Schedule 480 Volt for Busses F, G, H" circuits H19P00 through H19P12 and G25P00 through G25P13. The inspector observed that these drawings describe the power, control, and interlock circuits for the subject valves. The allegation was not substantiated.

(b) Allegedly no one knew how these circuits were routed in the plant. Licensee project engineering personnel stated that in addition to the drawings described above, the raceway schedule depicts circuits in a particular conduit, the conduit drawings show conduit locations in the plant, and the circuit schedule itemizes the pull data for each wire in the plant. They also stated that the drawings and schedules were available to the plant staff through the site document control center if this material was not available in the control room. The inspector had previously verified that this type of documentation was properly controlled and readily available to the plant staff. This allegation was not substantiated.

(c) It was alleged that the design was no good in that the control/interlock circuits are routed from the "hagen" racks via the solid state protection system to the relays which shut the valves. Licensee engineers explained that this was a standard Westinghouse design and that the "hagen" racks took low level analogue signals and (in this case) used bistables to

generate signals in the milliamp range. 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. They explained that they were not in a position to change this arrangement (since it is a Westinghouse design) and that they were unaware of any problems with this arrangement. The inspector examined the location of the components of the RHR isolation valve control and interlock circuits to verify the licensee's statements. The allegation was substantiated to the extent that the circuits were as alleged, however there was no apparent deviation from regulatory requirements or safety criteria.

(d) It was alleged that a design change request (DCR) submitted about February 1981 to get "rid of that system" (i.e. RHR hot leg suction isolation interlocks) has never been acted upon by PG&E. The inspector verified that there were no outstanding DCRs on PG&E drawing 437592 (which depicts the system in question) and that none were originated from or arrived at the Diablo Canyon project. The site Resident Inspectors verified that no DCRs were outstanding for this drawing at the jobsite. This allegation could not be substantiated.

(e) It was alleged that the FSAR, Chapter 5, paragraph 5.7, pages 37b and 38 as well as Chapter 7, paragraph 6.2, pages 3 and 4 describe the automatic high pressure/high temperature isolation of the RHR system from the reactor coolant system, and that this is inconsistent with the technical specifications section 3.4.9.3 which requires AC to be removed from the associated valves (8701 and 8702) thereby disabling the automatic isolation features. Therefore the FSAR should be amended. Licensee representatives showed the inspector Table 6.3-10 of the FSAR which shows that the valves are to be shut and racked out at power and open and racked out during shutdown cooling mode. This is in accordance with NRC direction. The licensee representatives also stated that the entire FSAR would be updated (with inconsistencies removed) in September 1983 in accordance with 10 CFR 50. The allegation was partially substantiated, but no safety problem or noncompliance with regulatory requirements was identified.

(f) The alleger stated that the FSAR section 3.1.3 states that spurious closure of normally open/fail open valves is not considered as either a passive or active failure and is not analyzed for at all which is a problem. Licensee engineers explained that there were no reasonable failure modes which would cause normally open/fail open or normally closed/fail closed valves to change state. The only possibility they could imagine was a "copper octopus" which caused selective shorting. This issue had been dealt with in the Fire Protection Review and was one reason that certain valve circuit breakers were racked out after the valve was placed in the desired position. As far as control circuits are concerned, any short with 120 volts or higher would cause the logic circuits to go to a fail safe condition due to the overwhelming signal strength (normal signals are 4 to 20 milliamps). The allegation could not be substantiated.

(g) It was alleged that there was no low flow alarm for the RHR system and that there should be one. The inspector verified that an RHR pump trip is annunciated, that shut RHR suction valves are indicated, and that the subcooling meter was available to ensure adequate core cooling. Licensee representatives pointed out that the RHR pumps have a miniflow recirculation to maintain some flow, and that the monitor light box indicates valves or circuits in the incorrect state. The inspector concluded that the allegation was correct in that there was no "low flow" alarm, but also concluded that there appeared to be no requirement or necessity to have one.

(h) It was alleged that an RHR pump ran without flow for 5 minutes in September 1981, and that this event was not reported as required by administrative procedure C-12 and 10 CFR 50.72. The site resident inspector verified that a Nuclear Plant Problem Report (DCI-81-OP P1057) and the associated corrective action was completed. The allegation was not substantiated.

(i) It was alleged that the RHR hot leg suction does not meet the single failure criteria for function (suction from reactor coolant system hot leg), that newer plants had this feature, and that this portion of the system should be redundant to meet 10 CFR 50 Appendix A Design Criteria. The inspector verified that this function was not safety related in the Diablo Canyon plant design by examining the FSAR. The inspector observed that the suction from the containment sump and from the refueling water storage tank were both safety related and arranged to meet regulatory requirements for redundancy. The inspector also observed that some other plants did have two RHR suction lines but that these plants used a different nuclear steam supply system vendor. The inspector concluded that the allegation was correct in that the RHR suction line was redundant only for the purpose of reactor coolant system isolation, but that there was no apparent safety problem or deviation from regulatory requirements associated with this design.

(j) It was alleged that nuclear plant problem reports (NPPR) were not getting management review which is a violation of administrative procedure C-12 and that NPPR DC 1-81-OP P1057 had been signed off after this shortcoming was identified to management. Other NPPRs should be examined. The Resident Inspectors observed that other NPPRs were being given appropriate management review and resolution. The allegation was not substantiated.

(k) It was alleged that NPPRs DCO 79 TI P0006 and 79 TI P0117 are still open after three years and should be closed. The Resident Inspectors observed that response to NPPR P0006 was complete and that response to P0117 was underway. The allegation was substantiated, but no particular safety or regulatory significance could be attached to this situation.

(l) It was alleged that a change to the Plant Manual Volume 16, reactor coolant pump "lo oil level" alarm should have been changed to "lo-hi oil level" but had not been corrected eight months after the correction had been submitted. The Resident Inspectors identified this allegation to the licensee. The licensee initiated a NPPR (DCI-83-TN-P0001) and the problem is to be resolved. The licensee personnel that were interviewed, were not previously aware of this problem. The allegation was substantiated.

The inspector concluded that the allegations were partially correct but that these had no apparent safety significance or deviations from regulatory requirements.

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