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DOCDATE: 01/26/78
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SUBJECT:
TRANSMITTAL OF INFO CONCERNING REVIEW OF SYSTEMS AND EQUIPMENT NECESSARY TO ACCOMPLISH A SAFE SHUTDOWN FOLLOWING A MAJOR EARTHQUAKE IN RESPONSE TO NRC LTR DTD 12/12/77.

PLANT NAME: DIABLO CANYON - UNIT 1
DIABLO CANYON - UNIT 2

REVIEWER INITIAL: XEF
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January 26, 1978

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Mr. John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Docket No. 50-275-OL
Docket No. 50-323-OL
Diablo Canyon Units 1 & 2

Dear Mr. Stolz:

The attached information is submitted in response to your letter of December 12, 1977 requesting a review of systems and equipment necessary to accomplish a safe shutdown following a major earthquake.

Kindly acknowledge receipt of this material on the enclosed copy of this letter and return it to me in the enclosed addressed envelope.

Very truly yours,

Philip A. Crane, Jr.

Attachments (40)
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Philip A. Crane, Jr.

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This is in response to a request for additional information concerning re-evaluation of the seismic design which was enclosed in Mr. Stoltz' letter of December 12, 1977.

The first question in the enclosure asks for a detailed description of the procedures proposed following the postulated Hosgri earthquake. The meeting held with the NRC staff in Bethesda on November 22, 1977, was concerned with this description. At that time, schematic diagrams of the plant systems involved in plant shutdown were used as the basis of a detailed explanation of the equipment, valves, lines, power sources, and instrumentation used for plant shutdown. We believe that this meeting provided the basic answer to Question 1. The question has four subsections to which we respond as follows:

a. We agree that systems or equipment proposed to be used in shutdown following the postulated Hosgri seismic event should be qualified seismically for that event. This is one of the basic assumptions behind the Hosgri Seismic Evaluation. All such systems and equipment have been qualified or are being qualified.

b. This asks about the effect which failure of systems, structures, or components not qualified (seismically) might have on the ability to shutdown. Since all structures are or are being qualified and all systems and components which are used to shutdown are or are being qualified, this limits the effect of the subject failures to possible consequential damage to be inflicted upon qualified systems. The measures taken to protect against damage subsequent to pipe breaks have been discussed in Section 3.6 of the FSAR. That discussion applies to the present case, because the jet and impingement forces are much greater than the seismic forces. Possible flooding is discussed in Appendix 3.6 to Section 3.6 of the FSAR and also in the safety evaluations of specific components. Reference to these sections will show that safe shutdown is not compromised by the failures postulated. We are reviewing all non-Class I lines in the containment to confirm that the postulated failure of any such line would not interfere with safe shutdown. We know of no case of such potential interference , but should one be identified during our review, modifications will be made to eliminate the condition.

c. The answer to this question is the attached Single Failure Analysis of Plant Functions Necessary to Achieve Cold Shutdown. Unless a single failure occurs, there is no need for an operator to leave the control room in carrying out the shutdown described in Section 5.1 of the Hosgri Seismic Evaluation, Amendment 50.

d. We agree that the loss of offsite power is a reasonable condition to postulate as following the Hosgri seismic event. This is one of the basic assumptions in the Hosgri Seismic Evaluation. The discussion of shutdown procedures in that document, in the meeting with the staff on November 22, and in the single failure analysis referred to above, is on the basis of loss of offsite power.

The second question in the enclosure to Mr. Stoltz' letter asks for a list of systems and equipment used in the shutdown procedure which was the subject of question one. The systems involved and the equipment required are listed in Section 7.3 of the Hosgri Seismic Evaluation. Some additional components were agreed upon in discussions with the NRC staff and appear in the minutes of the November 22 meeting. The various types of information listed in the question were discussed with the staff on November 22 and appear in the appropriate sections of the Hosgri Seismic Evaluation, with two exceptions.

1) Piping schematics may be found in Section 3.2 of the FSAR. A limited number of schematics marked by hand in color to indicate only the minimum flow paths required for shutdown following a Hosgri event have been given to the appropriate staff reviewers on two occasions. The sheets involved have been listed in Enclosure 2 of Mr. Israel's memo of December 8, 1977, to Mr. Bosnak.

2) Code classes of equipment are listed in Table 3.2-4 of the FSAR. All of the components required are seismic Design Class I with the exception of the steam generator 10% atmospheric steam dump valves which are Class II but which have been qualified seismically.

Question 3 asks that justification be provided for any approaches not used in normal shutdown (e.g., boration without letdown, boration with natural circulation) or recommended tests be described.

Boration without letdown is justified by calculations which have been made assuming full power conditions prior to shutdown, which have determined the volumes required to accomplish boration and depressurization.

In performing the calculations, the following plant operation was assumed following the trip. First, the operators would borate the RCS to the cold shutdown concentration, taking advantage of the steam space available in the pressurizer. Second, the operators would use the steam generators to cool the RCS to 350°F and would take advantage of the cooldown contraction to lower the pressurizer water level. Finally, the operators would use auxiliary spray from the CVCS to depressurize the plant to 425 psia.

The assumed initial conditions following plant trip are:

RCS Temperature = 547°F

RCS Pressure = 2250 psia

Pressurizer Water Volume = 500 ft^3

Pressurizer Steam Volume = 1300 ft^3

The volume of water is less and the volume of steam is greater than at full load, since some shrinkage of the RCS inventory will have occurred because of the reduction in temperature to no load conditions. To calculate the volume required for worst case conditions of end of core life and maximum peak xenon were assumed. This results in a requirement for 600 cubic feet of 12wt% boric acid at 165°F to reach cold shutdown concentration. When added to the RCS, the boric acid would be heated to 547°F and would expand to 800 cubic feet. Since this volume is less than the 1300 cubic feet available in the pressurizer steam space, boration to cold shutdown concentration can be accomplished without letdown, without taking the plant water solid, and without cooling down.

The cooldown from 547°F to 350°F decreases the volume of water in the RCS by approximately 1730 cubic feet. Some of this contraction is used to return the pressurizer water level to the no load water level after boration and the remainder is compensated for by charging to the RCS.

To calculate the volume required for depressurization, it was assumed that the pressurizer was at saturated conditions with 500 cubic feet of water, 1300 cubic feet of steam, and the pressurizer metal all at 653°F (2250 psia). It was further assumed that no additional water would be removed from the pressurizer by the cooldown contraction. With these assumptions, and including the effect of heat input from this pressurizer metal, it was determined that spraying approximately 820 cubic feet of 165°F water would produce saturated conditions at 425 psia (450°F) with a water volume of 1550 cubic feet and a steam volume of 250 cubic feet. These calculations demonstrate that boration and depressurization can be accomplished without letdown, without taking the plant water solid, and without taking full credit for the available volume created by the cooldown contraction.

Boration with natural circulation will be demonstrated by a boron natural circulation mixing test which will be performed at the plant. Because this demonstration requires residual heat in the core, it will be performed after fuel loading and criticality has been reached. The cooling rate of the reactor vessel head will be determined as a part of this test.

Shutdown without the removal of dissolved gasses from the reactor coolant system is justified by the following discussion. In order for a two phase system (vapor-liquid) to be sustained, it is necessary that the total pressure in both phases be equal; the vapor phase pressure will be due to both steam and non-condensible gasses and they must be in equilibrium with the liquid phase. The saturated (equilibrium) steam pressure for any given temperature can be obtained from the steam tables. At the point of RHR initiation (350°F, 425 psia) this pressure is 135 psia.

Hydrogen is essentially the only non-condensible gas present in the reactor coolant. Equilibrium hydrogen distributions between liquid and vapor phases have been empirically determined for many different temperatures. T. S. Anderson reviewed and evaluated these data with the results presented in WAPD-TM-6333, "Correlation of Solubility Data for Hydrogen and Nitrogen in Water," October 1967. For 350°F, this paper indicates a Henry's Law constant of 0.47 psia/(cc/kg)$_{STP}$. Henry's Law states:

$$P_i = H_i X_i$$

Where

P = partial pressure of constituent i.

H_i = Henry's Law constant.

X_i = liquid phase concentration of constituent i.

Thus, if the hydrogen pressure is to be equal to the difference between the total pressure and the steam pressure (i.e., 290 psi), the liquid phase hydrogen concentration must be approximately 600 cc/kg. At concentrations less than this, a two phase system will not be sustained.

The specified reactor coolant hydrogen limits during operation are 25 to 50 cc/kg. During operation, some of this hydrogen is transferred to the pressurizer where a vapor-liquid equilibrium is established. The magnitude of this hydrogen has previously been estimated due to concern about the presence of combustible gasses in containment following an accident. Using conservative assumptions, the draft of ANSI Standard 275 predicts 1.2 SCF of hydrogen per cubic feet of pressurizer vapor space for a reactor coolant hydrogen concentration of 35 cc/kg. Increasing this for operation at 50 cc/kg and assuming that all of the hydrogen is redissolved in the coolant uniformly results in an estimated additional concentration of less than 150 cc/kg. It should be noted, however, that this is a conservative result and would not be expected during normal operation. These two hydrogen sources would indicate a maximum hydrogen concentration of 200 cc/kg, well below the amount required to sustain a two phase system.

Hydrogen is added to the reactor coolant during normal operation to suppress the production of oxidizing species from the radiolysis of water. This is accomplished by forcing the recombination reaction in accordance with Le Chatelier's Principle. The magnitude of the hydrogen concentration required to eliminate the net decomposition or disassociation of water is dependent upon the strength of the radiation source. Paul Cohen, in Chapter 4 of his book (Water Coolant Technology of Power Reactors, an AEC Monograph, 1969) indicates that approximately 15 cc/kg hydrogen will eliminate water disassociation at power. The magnitude required for suppression during shutdown conditions will be significantly lower than this value. Thus, it can be concluded that if the hydrogen concentration exceeds 15 cc/kg, there will be no net radiolytic hydrogen production.

In conclusion, the following statements can be made:

1. Hydrogen at the maximum specified level (50 cc/kg) will not be sufficient in content to sustain a two phase system at RHR initiation conditions.

2. Radiolytic hydrogen production will not occur when the reactor coolant hydrogen concentration is greater than 15 cc/kg, which is below the specified level. Thus, this source does not enter into the evaluation of this problem.

Question 4 concerns the probability of mechanical failure of either of two valves (8701 and 8702) in the residual heat removal system piping. The mechanical failure of the disc separating from the stem has been investigated (WCAP-9207) and has been found to be in the range of 10^{-4} to 10^{-3} per year. The probability of a Hosgri earthquake is less than $2x10^{-5}$ per year. The combined probability of valve stem failure coincident with the earthquake ($<10^{-7}$ per year) is so low that it need not be considered in the single failure analysis. The stresses and accelerations on these valves have been evaluated for the postulated Hosgri event and have been found to be within the design allowable values for the expected range of service conditions. Thus, the experience which formed the basis for the statistics in WCAP-9207 is applicable to this case.

Mr. Stoltz' letter of December 12, 1977, refers to a previous letter of November 10, 1977, which requested information concerning sources of water for long term cooling. We would expect that long term cooling would be performed by the residual heat removal system which has been provided for this purpose and has been qualified for Hosgri seismic accelerations. The question of November 10, 1977, implies that for some postulated, but unidentified reason, it is necessary to remain at hot standby for an extended period of time. The basic source of water for this purpose, the condensate storage tank, will be sufficient for a number of hours, the exact length of time depending on the conditions assumed for the calculation. If a longer period of time is postulated, water can be provided from the raw water storage reservoir.

The raw water reservoir is excavated in a rock terrace east of and above the plant. The geology and geomorphology of the natural slopes in the area of the reservoir indicate that the slopes are stable and have been so for thousands of years. The strike of the rock is approximately northwest and is essentially perpendicular to the slope and the beds are predominantly steeply dipping to the northeast. The very favorable orientation of the resistant rock strata makes slope instability

very unlikely. The slopes have been subjected to seismic vibrations during their geologic history and have demonstrated that they are basically stable. Furthermore, the reservoir is lined with a flexible membrane which would prevent leakage of water out of the reservoir even if it were assumed that an earthquake could cause cracking of the concrete in the reservoir.

The reservoir is divided into two sections which communicate but which can be isolated. Both sections have piping connections to the plant. The piping is asbestos cement and was not intended to be earthquake resistant. These lines have valving at both ends and can be isolated if damaged. If a pipe line should suffer damage in an earthquake, the line would be isolated and may be replaced by fire hose. A hose line run after a seismic event would be immune to further seismic disturbance, could be adapted to any topography resulting from the event, and would be the fastest replacement that could be installed. A hose station will be established near the reservoir. Because deployment would be downhill, the labor involved would be minimized. Standard 2½ inch fire hose will be used because it is relatively easy to deploy, is compatible with the other plant fire hose, and will supply at least 200 gpm. Ten thousand gph is required per unit. Redundant valved connections will be installed in the piping at the plant to receive the hose line(s) from the reservoir. This will be at the same functional location where the present lines enter the plant makeup water system. Two million gallons of water will be held in the reservoir and will be available for use for this purpose. This is sufficient to maintain one unit at hot standby without offsite power for approximately eight days or two units for four days each.

As an ultimate backup source to provide water for an indefinitely extended period of hot shutdown, provision will be made to allow seawater to be used as auxiliary feedwater. Temporary connections would be made to the plant makeup water system. Redundant pumps will be provided which would deliver saltwater to be used for auxiliary feedwater pumps suction. Since the steam generators hold sufficient water for a number of hours of operation at hot standby, there would be time available to make the arrangements to change the source of water.

SINGLE FAILURE ANALYSIS OF PLANT FUNCTIONS NECESSARY TO ACHIEVE COLD SHUTDOWN

I. CIRCULATION OF THE REACTOR COOLANT

Natural circulation flow, with the reactor core as the heat source and the steam generators as the heat sink, provides adequate circulation of the reactor coolant during the first stage of cooldown from hot standby to 350°F. Four reactor coolant loops and steam generators are provided, any one of which can provide sufficient natural circulation flow to provide adequate core cooling. Under any single failure during plant cooldown, three reactor coolant loops and steam generators remain available. The effected steam generator could be made available by operator action to open the steam generator 10% atmospheric steam dump valve manually. During the second stage of cooldown, from 350°F to 200°F, the redundant RHR pumps each provide adequate circulation of the reactor coolant.

II. HEAT REMOVAL

A. During this stage, the steam generators are used for heat removal from the reactor coolant system, with steam release to the atmosphere and makeup to the steam generator from the auxiliary feedwater system.

1. Steam release to atmosphere is through the steam generator 10% atmospheric steam dump valves, one of which is provided for each of the four steam generators. In case of a single failure, three steam dump valves remain available. Only one is required. A steam dump valve which fails to open could be opened manually by operator action.

2. The primary source of auxiliary feedwater is from the condensate storage tank. However, to ensure the capability to operate for extended periods at hot standby conditions, 1,000,000 gallons of water are available per unit in the raw water reservoir to provide makeup to the auxiliary feedwater system. This quantity of water is sufficient to permit both units to remain at hot standby for 100 hours or to permit one unit to remain at hot standby for 200 hours, following shutdown from full power. Finally, design provisions allow sea water from the auxiliary salt water system to be provided as auxiliary feedwater makeup if necessary.

3. The first active components in the auxiliary feedwater flow path from the condensate storage tank to the steam generators are the auxiliary feedwater pumps. One turbine driven pump and two motor driven pumps are provided, any one of which can provide sufficient auxiliary feedwater flow. The two motor driven pumps are powered from different emergency power trains, and the turbine driven pump can receive motive steam from either of two steam generators. This system can withstand the single failure of any one pump, valve, or power supply and still supply water to all four steam generators.

The turbine driven pump is normally aligned to provide flow to all four steam generators via four normally open motor operated valves (LCV-106, 107, 108, 109). If any one of these valves spuriously failed closed, flow could still be provided from the turbine driven auxiliary feedwater pump to the remaining three steam generators. Each motor driven pump is normally aligned to provide flow to two of the four steam generators through normally closed electro-hydraulic valves (LCV-110, 111 and LCV-113, 115) which are powered from emergency power trains. Normally closed manual cross-connect valves (in line K16/4292/4) can be opened to allow either motor driven pump to feed all four steam generators.

B. From 350°F to 200°F

1. Residual Heat Removal System

a. Suction Isolation Valves 8701 and 8702

These valves are powered from different emergency power trains. Failure of either power train or of either valve operator could prevent initiation of RHR cooling in the normal manner from the control room. In the event of such a failure, operator action could be taken to open the effected valve manually. The mechanical failure of the disc separating from the stem has been investigated (WCAP-9207) and has been found to be in the range of 10^{-4} to 10^{-3} per year. The probability of a Hosgri earthquake is less than $2x10^{-5}$ per year. The combined probability of valve stem failure coincident with the earthquake ($<10^{-7}$ per year) is so low that it need not be considered in the single failure analysis. The stresses and accelerations on these valves have been evaluated for the postulated Hosgri event and have been found to be within the design allowable values for the expected range of service conditions. Thus, the experience which formed the basis for the statistics in WCAP-9207 are directly applicable.

b. Isolation Valves 8700A and 8700B

If either of these normally open motor operated valves, which are powered from different emergency power trains, were to close spuriously, RHR cooling would be provided by the uneffected RHR pump and heat exchanger. The effected valve could be opened by operator action manually.

c. RHR Pumps 1 and 2

Each pump is powered from a different emergency power train. In the event of single failure, either pump provides sufficient RHR flow.

d. RHR Heat Exchangers 1 and 2

If either heat exchanger is unavailable for any reason, the remaining heat exchanger provides sufficient heat removal capability.

e. RHR Flow Control Valves HCV 637 and 638

If either of these normally open, fail open valves should close spuriously, sufficient RHR cooling would be provided by the unaffected RHR train.

f. RHR/SIS Cold Leg Isolation Valves 8809A and 8809B

If either of these nomally open, motor operated valves, which are powered from different emergency power trains, should close spuriously, sufficient RHR cooling would be provided by the unaffected RHR train. The effected valve could be opened by operator action manually.

2. Component Cooling Water System

a. Component Cooling Water Pumps 1 and 3

Each pump is powered from a different emergency power train. In the event of a single failure, either pump provides sufficient CCW flow.

b. CCW Heat Exchangers 1 and 2

If either heat exchanger is unavailable for any reason, the remaining heat exchanger provides sufficient heat removal capability.

c. CCW Isolation Valves FCV-430 and 431

These valves are powered from different emergency power trains. One of them is open during normal operation. If it has not closed spuriously, neither of the valves is required to operate.

3. Auxiliary Salt Water System

a. Auxiliary Salt Water Pumps 1 and 2

Each pump is powered from a different emergency power train. In the event of a single failure, either pump provides sufficient ASW flow.

b. Isolation Valves FCV 602 and FCV 603

These valves have redundant control power. One of them is open during normal operation. If it has not closed spuriously, neither of the valves is required to operate. These valves are air operated and open on loss of air.

III. BORATION AND MAKEUP

A. Boration

1. Boric Acid Tanks 1 and 2

Two boric acid tanks are provided. Each tank contains sufficient 12 wt.% boric acid to borate the reactor coolant system for cold shutdown conditions. Each tank is provided with redundant heaters powered from different emergency power trains. Redundant heat tracing circuits are provided on all lines which normally contain 12 wt.% boric acid.

2. Boric Acid Pumps 1 and 2

Each pump is powered from a different emergency power train. In the event of a single failure, either pump will provide sufficient boric acid flow.

3. Isolation Valve 8104

If valve 8104, which is supplied from emergency power and is normally closed, does not operate, it can be opened manually by operator action. An alternate flow path is available through air-operated valve FCV-110A which fails open on loss of air and manually operated valve 8471 which is normally closed. In addition, if air has not been lost, the normal path through the blender remains available.

B. Makeup

The refueling water storage tank provides a source of makeup in addition to that provided as 12 wt.% boric acid for boration.

1. Isolation Valves 8805A and 8805B

Powered from different emergency power trains, only one of these two valves needs to open to provide adequate makeup flow.

C. Boration and Makeup

Four different boration paths to the RCS are available. These are: 1) the normal charging header; 2) the reactor coolant pump seals; 3) the alternate charging header; and 4) the auxiliary spray header.

1. Centrifugal Charging Pumps 1 and 2

Each pump is powered from a different emergency power train. In the event of a single failure, either pump provides sufficient boration or makeup flow.

2. Flow Control Valve FCV-128

This normally open valve fails open on loss of electrical power or air. If FCV-128 should close spuriously, the charging pumps would operate satisfactorily on their mini-flow circuits and operator action could be used to open bypass valves 8387B and 8387C. Flow indication, qualified for the Hosgri earthquake, will confirm that valve FCV-128 is open.

3. Flow Control Valve HCV-142

The normally open, fail closed valve, is provided with an air backup nitrogen supply. If HCV-142 should close spuriously, operator action can open manual bypass valve 8403. In addition, the flow path via the reactor coolant pump seals is still available.

4. Isolation Valves 8107 and 8108

If either of these normally open, motor operated valves, which are powered from different emergency power trains, should close spuriously, operator action could be used to de-energize the valve operator and reopen the valve with its manual handwheel. In addition, the flow path via the reactor coolant pump seals is available.

5. Isolation Valve 8146

If this normally open, fail open, valve should close spuriously, alternate charging header valve 8147 or auxiliary spray valve 8145 could be opened. In addition, the flow path via the reactor coolant pump seals is available.

IV. DEPRESSURIZATION

Depressurization is accomplished by auxiliary pressurizer spray from the CVCS. Either 12 wt.% boric acid or makeup from the RWST can be used as a source for auxiliary spray. The only portion of the depressurization path which is different from that discussed for boration and makeup is the auxiliary spray valve.

A. Auxiliary Spray Valve 8145

This normally closed, fail closed, valve is provided with an air backup nitrogen supply. In the event 8145 was not operable, auxiliary spray for depressurization can be provided by operator action to open a manual bypass valve around 8145.