BEFORE THE UNITED STATES ATOMIC ENERGY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY	Docket No. 50-205 Amendment No. 2

Now comes PACIFIC GAS AND ELECTRIC COMPANY (the Company) and amends its above-numbered application by submitting herewith Amendment No. 2, which sets forth in question and answer form additional information concerning Unit No. 1 of the Company's Bodega Bay Atomic Park.

Subscribed in San Francisco, California, this 5th day of April, 1963.

Respectfully submitted,

PACIFIC GAS AND ELECTRIC COMPANY

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U.S. Atomic Energy Commission
DIVISION OF LICENSING & REGULATION

Subscribed and sworn to before me this 5th day of April, 1963

Rita J. Green (SEAL)
Rita J. Green, Notary Public in and for the City and County of San Francisco, State of California
My Commission expires July 16, 1963.

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AMENDMENT NO. 2
DOCKET 50-205
PACIFIC GAS AND ELECTRIC COMPANY
BODEGA BAY ATOMIC PARK-UNIT NO. 1

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1. Discuss dry well and suppression chamber integrity tests.

   The design pressure for the dry well will be 62 psig; and for the suppression chamber, 35 psig. The initial tests will include:

   (a) A soap bubble test at 5 psig, applied to all welds and seals.

   (b) An ASME pneumatic pressure test at 115% of design pressure, with penetrations and vent pipes capped.

   (c) Leak rate tests at design pressure with penetrations and vent pipes capped. (Specifications for this test will require that leakage not exceed 1/10 of 1% of the volume per day).

   (d) Leak rate tests at reduced pressure with penetrations installed and concrete in place (with vent pipes still capped) to demonstrate that leakage rates from dry well and suppression chamber do not exceed rates equivalent to 1/2 of 1% per day at design pressure.

   Leak rate tests after the plant is placed in operation will be determined later. A continuous leak rate detection system will be installed if such a system proves successful at Humboldt.

2. Give further information on the dry well cooling system.

   The dry well is cooled by four heat exchanger and fan units, as shown in Figure III-3 of the Preliminary Hazards Summary Report (PHSR). Air is recirculated within the dry well and the cooling system is designed to limit the air temperature to 150°F. Dry well penetrations are provided for cooling water to the heat exchangers, and for electrical and control connections.

3. Discuss thermal expansion problems in the pressure suppression containment system and give the containment design criteria.

   The normal environment in the dry well during plant operation is essentially atmospheric pressure and an average temperature of 150°F. The suppression chamber is at atmospheric pressure and a temperature in the range of 60 to 90°F. The dry well and the [illegible] system will be designed for a pressure of 62 psig. The suppression chamber toroid will be designed for a pressure of 35 psig.

   Thermal stresses in the steel shells due to temperature increases will be provided for in the design of the vessels. Since the Maximum Credible Operating Accident (MCOA) peak dry well pressure rapidly diminishes to 35 psig or less, the design dry well temperature to be considered along with 62 psig pressure will be the saturation temperature of steam at 35 psig, or about 280°F. The maximum suppression pool temperature following the MCOA will be 140°F.

In addition to thermal stresses and internal pressures, the pressure suppression system will be designed for all other loads and forces to be sustained, including the attached equipment and structures. All structures will be designed to withstand vertical and lateral forces due to seismic shock. Design criteria for seismic shock shall be as given on page V-8 of the PHSR. Design, fabrication, erection and testing of the pressure vessels and interconnecting piping will be in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code (latest edition) and the ASME Code Cases applicable to nuclear containment vessels.

Because of the foundation and earthquake design requirements for the reactor containment system, which is classified as a Class 1 structure, it will be necessary to embed at least some portions of the containment vessels in concrete. Protection against thermal stresses can be provided by one or more of the following methods: (1) provision of a flexible plastic material of predetermined thickness between the steel shell and the surrounding reinforced concrete to provide for expansion, contraction or deflection; (2) provision of a suitable insulating material within the dry well containment to minimize temperature effects; (3) provision of suitable expansion joints in the cylindrical portion of the dry well vessel as required to minimize thermal expansion stresses.

It is estimated that temperature increases in the order of 60 to 75°F above ambient can be safely withstood by the steel shells when rigidly embedded in concrete and exposed to the full temperature increase. Special stress analyses and design provisions will be made for the relatively short period when additional temperature increases due to accident conditions could exist.

The final details of embedment of the vessels, and special design provisions to sustain thermal stresses, earthquake shocks and other loads will be determined after detailed study by the Company and the contractor.

4. Describe the refueling building ventilation system, and its function during building isolation.

Bodega will use essentially the same type of refueling building ventilation system as has been provided for Humboldt Bay. Normal ventilation will be directly to the stack. At a high radiation signal, the refueling building will be isolated and ventilation will be only by way of the gas treatment system.

Reliability of this system is assured by the provision of a number of monitoring instruments and of two complete sets of gas treatment equipment. Either of these sets can be placed on emergency power on loss of normal power.

5. What is the allowed leak rate of the safety relief valves in the main steam line and what is the effect of leakage on dry well pressure?

Valves with a maximum leak rate per valve of 0.3 scf/day of air at 90% of set pressure will be provided. (This is approximately equivalent to 0.001 lb/hr of steam per valve). Seat leakage will be vented to the

suppression pool, where the steam will be condensed. A small quantity of entrained gases could reach the suppression chamber and, by way of the vacuum breakers, the dry well. Any increase in pool temperature or dry well pressure or temperature as a result of valve leakage would be detected. The Technical Specifications will include appropriate operating limits for pool temperature and dry well pressure and temperature.

6. Do the dry well closures seal automatically on high dry well pressure?

Normally, the dry well will be sealed during operation. Those closures which may occasionally be opened by operator action in the control room will be closed and sealed automatically by an overriding high dry well pressure isolation signal. Examples of such closures are those in the dry well purge and dry well drain lines.

7. What measures will be taken to assure early discovery of pipe failures in the dry well and in other inaccessible spaces?

Large leaks will be detected by temperature and/or pressure sensors which will alarm and close the appropriate isolation valves. Small leaks will be detected by temperature sensors and by liquid level buildup in sumps. It is estimated that a 300 lb per hour steam leak in the dry well would be apparent from temperature rise, and that observation of liquid level buildup in the dry well sump would make it possible to detect even lower leakage rates.

8. Would the ducting of steam from the relief valves in the main steam line alter the peak suppression chamber pressures as seen in the Moss Landing tests?

Ducting of steam to the suppression pool before the MCOA would increase the pool temperature; but the Moss Landing tests demonstrate that pool temperature has a negligible effect on peak suppression chamber pressure. These tests show that chamber pressure is determined largely by the amount of air which is transferred from the dry well to the chamber during blowdown of the reactor vessel.

9. Where are the isolation valves on the main steam line located?

The main steam line consists of two 20-inch pipes in parallel leading from the steam headers in the dry well to the turbine. Each pipe has two isolation valves: the first, inside the dry well and adjacent to the dry well wall; and the second, the turbine stop valve, in the turbine pedestal.

10. Discuss the criterion for setting the length of time required for closure of the isolation valves.

The criterion for closing time of the isolation valves is that, in the event of a gross break in a line outside the dry well, the valves will close in time such that the reactor core can be covered again by continued feedwater flow before decay heating can increase fuel temperatures to a level which will cause fuel clad failure. Analyses are in progress to determine clad temperature and stress levels versus time for various valve closure times following a main steam line break, and it is intended that the results will form the basis for the selection of valve closing time. As reported in Amendment No. 1, it is estimated that the required time will be between 10 and 30 seconds.

11. Discuss the effect of isolation valve location on the consequences of a nuclear incident.

All lines which open into the dry well or reactor system have double barrier protection (two valves) against leakage from the containment system in the event of an MCOA or other accident which would release gross quantities of fission products within that system because of core melting.

In some instances, one of the two isolation valves will be located some distance from the containment barrier. This is believed justified because no mechanism is foreseen by which a break in the line could occur coincident with a gross fission product release from the fuel. For example, in the event of a main steam line break the valve at the dry well barrier will close in time such that there will be no danger to the fuel from overheating (see answer to Question 10); and the fuel and core structure will be designed to prevent lifting or mechanical damage to fuel during the blowdown period while the valve is closing. Even if the valve should fail to close, the reactor would scram and core spray plus rapid refilling of the vessel should prevent damage to the fuel and thus there would not be a potential for fission product release.

12. Are there dead flow or low flow regions in the reactor vessel between the flow baffle and the vessel wall at the inlet plenum or between the vessel wall and the thermal shield where excessive heat may occur?

There are no such regions in the vessel. The design shown in Figure III-10 of the PHSR has been changed and does not contain a gamma shield adjacent to the reactor vessel wall. A gamma heating analysis indicates that the maximum reactor vessel shell temperature rise due to gamma heating will be less than 15°F. The maximum temperature rise occurs in the reactor vessel shell opposite the mid-plane of the core.

13. Discuss the effect of the choice of 6" steps for control rods as opposed to 3" steps for Humboldt Bay.

Calculations to determine step worths have not been completed and it may be that a different length will be chosen for the final design. The intent is to limit maximum step worths in normal operating patterns to about 0.003 Δ k, and preliminary estimates indicate this can be achieved with 6 inch steps. This is the same maximum normal pattern step worth as is specified for Humboldt Bay.

14. What is the anticipated pressure transient in the reactor vessel on turbine trip? How would this be affected if one or more of the turbine bypass valves should fail to open? How will failure to open or close be indicated?

An analysis of this transient shows that a peak pressure rise of 100 psi above the normal operating pressure of 1060 psig will occur following a turbine trip. The neutron flux peak resulting from the pressure increase will initiate reactor scram. Reactor pressure will then be controlled by the automatic operation of the eight bypass valves. If all bypass valves should fail to open, safety valves would operate and limit pressure to about 1200 psig until the heat rate fell to the isolation cooling system capacity. Peak neutron flux would be the same in either case and peak fuel rod surface heat flux would not exceed 120% of rated in this transient with or without the bypass valves operating.

Indicator lights in the control room will show operation of the bypass valves and temperature instrumentation on the bypass discharge line will indicate bypass steam flow. Manually initiated shut off valves in each line can be used to stop flow through a leaking bypass valve. Failure of a valve to reclose is unlikely, but calculations show that the reactor vessel would not experience excessively high stresses as a result of the blowdown if this should happen.

15. Describe the reactor water level instrumentation.

There will be four level indicators, two for the feedwater control system and two for safety system sensing. The two for the feedwater control system are differential pressure devices, using a reference leg connected to an overflow weir inside the reactor vessel which is normally kept full by drainage from the steam dryer. Level indicating devices for safety system sensing have not been selected, but will probably also be of the differential pressure type. Differential pressure type level indicators give accurate level information except possibly under extreme transient conditions as, for example, would occur in a gross steam line break accident while the vessel contents were flashing. In this case they would again give reliable indication after flashing was stopped by closure of the isolation valves.

16. Will the poison injection system be on the emergency power system? How will the system be activated and how will it be tested?

The liquid poison system consists of two separate injection pumps which are connected to the normal a-c power supply. These pumps can be connected to emergency power on loss of normal power. Operation of the

liquid poison system is manually initiated from the control room by operation of two switches; one switch operates the explosive type injection valves and the other starts pumping of the liquid poison. The liquid poison system components are arranged so that they can be tested by circulating sodium pentaborate solution from the poison storage tank or water from a test tank.

17. State the locations of the components of the shutdown cooling system and of the core spray pumps and cooler.

The locations of the principal components of the shutdown cooling system are as follows:

(a) Emergency condenser - In the refueling building above elevation 62'3", as shown in Figures III-2 and III-7 of the PHSR.

(b) Shutdown heat exchangers and shutdown pumps - In a vault under the refueling building at elevation -6'6", as shown in Figure III-4. This vault is outside the dry well and is accessible during operation.

(c) Suppression pool coolers - In a vault under the refueling building at elevation -6'6", as shown in Figure III-4. This vault also is outside the dry well.

(d) Suppression pool recirculation pumps - In a compartment at the lowest level under the dry well, taking suction from the suppression pool.

(e) Control rod drive pumps - In a vault adjacent to the turbine pedestal, as shown in Figure III-5. (Three pumps will be provided, although only two are shown).

(f) Start-up pump - On the turbine pedestal at elevation 62'3", as shown in Figure III-7.

(g) Core spray pumps - Adjacent to the suppression pool coolers, as described in (c), above.

18. Discuss the effect of small condenser leakage leading to increased chlorine content of the primary water.

Chloride intrusion by way of the main condenser will be guarded against by the following design features:

(a) The condenser tubes will be aluminum brass, which experience has shown to resist sea water attack.

(b) The tubes will be either rolled and welded to the tube sheet, or provided with a flexible gasket at the rolled joint, to minimize leakage.

(c) Baffles will be provided adjacent to each tube sheet and condensate from this region (where leaks are most likely to occur) will be

collected and passed over a conductivity cell before it reaches the hotwell. If the conductivity should be high, indicating a possible leak, this flow can be valved off.

(d) Conductivity cells will be located in the hotwell to provide an indication of possible contamination.

(e) All condensate will be demineralized before being returned to the reactor. Conductivity cells downstream from the demineralizer will be set to alarm if a breakthrough should occur.

19. In which systems is salt water cooling used?

Salt water is used in the turbine condenser, the auxiliary coolers, the evaporator brine heater, and in the suppression pool cooler if necessary for heat removal following an accident.

20. Is there a single controller for the turbine bypass valve?

The eight solenoid-operated turbine bypass valves are operated in sequence from switches positioned around a cam driven by a single mechano-hydraulic pressure controller mounted in the turbine standard. Remote manual operation of the valves is also possible from the control room.

21. Discuss the plan to design some primary circuit components-such as the turbine and feedwater systems-to 0.2G acceleration.

The design criteria on page V-8 of the PHSR have been revised to include primary components such as the turbine and feedwater systems as Class I structures.

22. Discuss the possibility that earthquake scrams should be provided which also close the related isolation valves.

It is the practice of the Pacific Gas and Electric Company to design all of its power plants with full consideration for potential earthquake hazards and with the basic requirement that the plants should remain in operation to provide electric power. The plants have operated through a number of earthquakes of varying degrees of intensity without tripping off the line and without damage to plants. A continuation of this practice for atomic plants is desirable.

It is unlikely that a high intensity anticipatory earthquake scram device could be developed. Use of a conventional earthquake scram device would not result in scram or valve closure until after a high intensity earthquake was over. If a need arises to close isolation valves during or after an earthquake shock, other scram devices incorporated in the plant will cause these valves to close.

23. If a turbine casing rupture should occur, and if this should cause a rupture in the feedwater line, what would be the consequences? What would be the consequences of a coincident rupture of a main steam line and the feedwater line outside the dry well?

Although a failure of this type is unlikely, it is conceivable that the failure of a turbine wheel in the LP section could cause the rupture of

the exhaust hood and also cause loss of main feed flow by damaging the feedwater heaters in the condenser. If this were to happen, loss of condenser vacuum would scram the reactor, initiate closure of the steam line isolation valves and turbine stop valves, and prevent operation of the bypass valves. As any steam leakage beyond the turbine admission valves would be limited by the capacity of these valves, closure of the isolation and/or stop valves would limit steam loss to a small enough quantity that the reactor core would still be covered after the valves were closed.

In the event of a simultaneous rupture of a steam line and the feedwater line outside the dry well, the reactor would be scrammed and the steam line isolation valves would be closed by a "main steam line break" signal. The check valve in the feedwater line would prevent significant loss of reactor water through this line. Loss of steam and water through the steam line would be such that the core would be partially or completely uncovered (depending on isolation valve closing time) before closure of the isolation valve stopped flow out of the system. The core would be recovered with water pumped by the startup feed pump through core spray system piping and by the rod drive feed pumps before major fuel damage could occur. Lines from these pumps could not be damaged by this accident as they are not in the main steam line tunnel.

24. Discuss the basis for omission of filters and iodine removal systems from the active gas disposal system.

The off-gas disposal system will incorporate a high efficiency filter for the removal of particulate matter. Installation of iodine removal equipment is not planned because experience with other boiling water reactors indicates that the iodine content of the off-gas will be considerably below concentrations that could safely be released out the stack. It is planned to propose a permissible emission rate for iodine that will consider possible reconcentration effects in the environs, and to provide stack sampling equipment for iodine to assure that releases do not exceed the permissible level. If results from stack and environs monitoring programs should indicate a danger of approaching permissible limits, iodine removal equipment would be installed, but experience with other plants shows that it is unlikely that this will be necessary.

25. Discuss the monitoring of the condenser discharge.

Condenser off-gases will be monitored with a continuous gross gamma monitor which will alarm if the instantaneous noble gas discharge rate reaches the permissible annual average discharge rate, and will initiate closure of the off-gas isolation valve if the level reaches ten times the permissible annual average rate. Cooling water discharge, after mixing with other plant effluent streams, is continuously sampled in the discharge tunnel and the samples are periodically analyzed to assure that permissible discharge limits have not been exceeded. It is not expected that the condenser discharge will contribute significantly to the total radioactivity of plant origin in the liquid discharge from

the plant because pressure differentials are such that the cooling water would leak into the condenser, rather than condensate leak into cooling water, if a condenser tube leak should develop.

26. Discuss the effect of halogen content of the primary system on the main pipe break outside the dry well with associated ground release.

The analysis of this accident given in the PHSR assumed the halogen content of the reactor water was 50 μc/gram at the time of the accident. This is the radioactivity limit that will be proposed for the plant's Technical Specifications and is several orders of magnitude greater than is expected during normal operation. The released steam was assumed to cool rapidly and form a hemispherical cloud at ground level. No credit was taken for subsequent atmospheric diffusion which would occur after cloud temperature reached ambient, and thus the analysis is conservative for any atmospheric diffusion condition. A wind speed of 10 MPH was assumedon the basis that for lower wind speeds, cloud cooling would be slower and the steam would rise in the air. Actually, the cloud would be expected to rise considerably even with a 10 MPH wind speed, and thus the analysis is conservative in this regard also. Thus, in the event of this accident, it is doubtful that any person would receive an inhalation dose to the thyroid as high as the 2 rem calculated in the PHSR.

27. What are the consequences of the MCOA, taking no credit for fallout in the dry well and taking credit for plate out only as provided in treatments of similar problems in TID 14844?

A preliminary answer to this question was given in Amendment No. 1. More refined calculations made since submitting Amendment No. 1 indicate that the answer given then was low.

If no credit is taken for fallout and only 50% credit is taken for plateout in the containment, as suggested by TID 14844, calculated maximum exposure rates would be increased to 0.008 and 0.04 rems to the thyroid per hour of exposure for the unstable and moderately stable cases, respectively, compared to the 0.0011 and 0.005 rems per hour of exposure given for these cases in the PHSR. Assuming no wind diversity for the duration of the accident, calculated integrated exposures would be 1.5 and 7 rems to the thyroid for the unstable and moderately stable cases, respectively, compared to the 0.019 and 0.09 rems reported in the PHSR.

As indicated in Amendment No. 1, the calculations given in the PHSR are based on estimated effective half lives for halogen plateout and fallout. The estimated removal half life for halogens in the dry well used in the PHSR calculations agrees well with Canadian and British experimental work reported in the literature and is believed to conservatively represent a phenomena which would most certainly occur if halogens were released to the dry well.

28. Assume a nuclear excursion violent enough to release fission products from the fuel occurs. Discuss the containment of the fission products reaching the condenser and off-gas holdup system and the possibility that the turbine exhaust hood diaphragms might be blown out as a result of the excursion.

The excursion would cause the reactor to scram on high flux. Released noble gas fission products would be swept out of the reactor with steam and would go through the turbine and condenser and on into the off-gas holdup piping where they would be trapped by automatic closure of the off-gas isolation valve before any reached the stack. Released halogen and solid fission products would be absorbed in the reactor water, but some could leave the reactor with water ejected in the excursion and go into the condenser hotwell. Subsequent release of fission products from the reactor vessel would be small in comparison to those released in the first few seconds. The steam line isolation valves would be closed by a low condenser vacuum or other automatic closure signal if not already closed by the operator. The condenser would gradually lose vacuum over a period of 10 minutes or more as a result of in-leakage through the turbine shaft seals. Operation of the gland seal exhausters would be continued after condenser vacuum was lost to maintain the turbine shaft seals under a slight vacuum and discharge any seal leakage flow to the stack. Emergency power is available so that operation of the exhausters could be continued if normal power were lost. The system would be held in this manner until the fission products decayed sufficiently to allow safe disposal. The mechanical vacuum pump, which can be valved to discharge via the refueling building gas cleanup system to the stack, provides an alternate means of holding the condenser under vacuum during the holding period.

It is not believed that there is a potential for blowing the exhaust hood diaphragms as a result of an excursion. Maximum equilibrium pressure in the reactor vessel as the result of a rapid energy addition of 2700 Mw-sec (energy release for the Rod Drop Accident described in the PHSR) would be about 1150 psia. Assuming an energy release this great occurred during power operation, the pressure rise would increase steam flow momentarily, but not beyond the capacity of the condenser to handle it without danger of rupture of the exhaust hood diaphragms. This pressure rise would increase condenser load momentarily by a factor of two which would raise condenser pressure from 1-1/2" Hg Abs to less than 5" Hg Abs. The fission gas release with the excursion would not increase condenser pressure significantly. The gas release as a result of fuel rod rupture and burnout in a 2700 Mw-sec excursion is calculated to be equal to the total fission gas content of 3% of the fuel rods in the core. If the ruptured rods all had 15,000 MWD/T exposure, the gas release would total no more than 45 gram moles or about 36 scf. Rapid release of this amount of gas into the condenser (20,000 ft^3 vapor space volume) would increase condenser pressure by 0.06 inch of Hg as compared to a pressure rise of greater than 30 inches of Hg before there would be any danger of diaphragm rupture.

The turbine is provided with low vacuum trips which scram the reactor and close the steam line isolation valves and turbine stop valves if

condenser pressure reaches 7" Hg absolute and which give a second scram signal and prevent bypass valve operation if condenser pressure reaches atmospheric. These would prevent blowing the diaphragms in the event the off-gas isolation valve were closed and the reactor continued to operate after an accident that released fission products but did not scram the reactor.

29. Discuss the plans for ecological surveys.

The ecological surveys now being carried out for Bodega Bay are separate and distinct from the proposed future environmental monitoring surveys for the site. The purpose of the ecological surveys in the marine environment is to establish a background level of types, quantities and locations of marine flora and fauna. The program for the ecological surveys has been established by Dr. Ernest Salo of Humboldt State College, who has been the marine consultant for Humboldt Bay, for the oceanographic studies at Bodega Bay and for the proposed environmental monitoring surveys for Bodega Bay. Dr. Salo and his staff made initial their reconnaissance of Bodega Bay in May of 1962 and field work commenced in July. Field work has continued under the direction of Dr. Salo.

The Bodega area is characterized by 3 ecological zones, (1) Bodega Harbor, (2) the outer coast, and (3) the near-shore waters. The program includes detailed studies of specific areas within these zones to determine the specific types and numbers of marine life existing within these areas at any time. As this program continues, changes or trends in types and population will be observed prior to plant operation and after plant operation commences. The ecological survey also includes a study of commercial and sports fishing in the Bodega Bay area. Quarterly reports will be issued as soon as the program is fully under way. It is anticipated that the ecological survey will extend through plant startup and for some time into the post-operational period.