BEFORE THE UNITED STATES ATOMIC ENERGY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY

Docket No. 50-205

Amendment No. 1

Now comes PACIFIC GAS AND ELECTRIC COMPANY (the Company) and amends its above-numbered application by submitting herewith the Company's answers to the questions raised by the Division of Licensing and Regulation and attached to the Commission's letter dated February 26, 1963.

Subscribed in San Francisco, California, this 4th day of March, 1963.

Respectfully submitted,

PACIFIC GAS AND ELECTRIC COMPANY

By N. R. Sutherland
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By Philip A. Crane, Jr.
Philip A. Crane, Jr.

Subscribed and sworn to before me this 4th day of March, 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.

PACIFIC GAS AND ELECTRIC COMPANY
BODEGA BAY ATOMIC PARK - UNIT NO. 1

ANSWERS TO QUESTIONS RAISED BY THE DIVISION OF LICENSING AND REGULATION RELATIVE TO CONSTRUCTION PERMIT APPLICATION

1. As required by Paragraph 50.35, Title 10 of the Code of Federal Regulations, (see revision effective January 29, 1963) describe the research and development programs which will be conducted to resolve those design problems having a bearing on the safety of the plant.

   The following research and development programs will be conducted by the Company:

   (a) Meteorology. A meteorological facility is being installed at the site to provide necessary data for atmospheric diffusion studies. Instruments will be mounted at three levels on a 250 ft. tower and will measure temperature and wind speed and direction. All readings will be digitized and recorded on paper tape. (PHSR, page V-9, and Appendix I).

   (b) Oceanography. The capacity of the ocean to diffuse the condenser cooling water and minimize the effects of temperature and radio-activity on the marine biota is being investigated in a series of tests conducted at the site. These tests include use of drift poles and uranine dye as well as measurements of temperature and salinity. They will continue through at least one annual cycle of oceanographic and meteorological conditions. (PHSR, page V-11)

   (c) Marine Biology Survey. An ecological survey is being conducted to prepare check lists of the marine fauna and flora of Bodega Head and Harbor. (PHSR, page V-12)

   (d) Radiological Survey. A preoperational monitoring survey of the site and its environs will be initiated two years before commencement of operation of the reactor. The details of this program have not been completed for Bodega Bay. However, it is anticipated that it will be similar to that conducted for the Company's Humboldt Bay nuclear unit. (PHSR, page V-13)

   (e) Pressure Suppression Tests. Additional tests will be conducted at the Company's Moss Landing Power Plant to determine whether or not baffles between vent pipes are required in the suppression pool.

In addition to the research and development work being carried out by the Company the General Electric Company is carrying out a number of research and development programs of safety significance that will influence the design of the Bodega Bay plant. These are:

(f) Fuel Development. Results from fuel element development tests and experience with fuel designs now employed in existing reactors will form the basis for the selection of the Bodega fuel. See answer to question 4.

(g) Instrumentation Development. In-core startup range neutron detectors are being developed as a possible substitute for the previously planned out-of-core detectors. See answer to question 23.

(h) Control System Development. A prototype Bodega control rod drive is currently being manufactured. It will be subjected to extensive developmental testing before the final drive design is released for manufacture. Several devices which could reduce the likelihood or magnitude of a control rod drop-out accident are being developed for possible use in the Bodega control system, as discussed in the answer to question 44.

(i) Nuclear Excursion Analysis Development. Analytical models are being developed for the more accurate prediction of the physical consequences of nuclear excursion as discussed in the answer to question 44.

2. Delineate the division of responsibility between Pacific Gas and Electric Company and General Electric Company, in this project.

   General Electric Company will design and furnish the nuclear steam supply system, and will specify design requirements for reactor auxiliary systems. Pacific Gas and Electric Company is responsible for the design of the overall plant and will supervise its construction. In addition, General Electric Company has been retained as nuclear consultant.

3. Indicate whether a filter will be provided for the air ejector discharge to the stack and the approximate delay time provided ahead of this filter. (Figure II-1).

   A particulate filter will be provided at the end of the air ejector off gas holdup line. Approximately twenty minutes of delay time will be provided by the pipe ahead of this filter.

4. Justify, through reference to previous operating experience and experiments, your fuel element design using stainless steel cladding having a thickness of 11 mils.

The 11 mil clad fuel described in the Preliminary Hazards Summary Report has been tentatively selected for the Bodega reactor. The final design will not be selected until early in 1964. Developmental fuel designs, including some with 11 mil clad, are presently undergoing testing at various reactor facilities as part of General Electric Company and Atomic Energy Commission development programs. Experience with these fuels and with standard fuel designs now employed in existing boiling water reactors will form the basis for selection of the fuel design to be used for the Bodega plant. If experience indicates that 11 mil stainless fuel would perform satisfactorily, it may be used. Alternate clads that might be used are stainless steel of greater thickness or zircaloy.

5. Indicate the height of water shielding which will be provided above a fuel element when it is being transferred through the gate opening from the reactor to the storage pool. State the calculated radiation dose rates at the pool surface under these conditions.

A minimum water cover of $8\frac{1}{2}$ feet will be provided above a single irradiated fuel element when it is being transferred from the reactor to the spent fuel storage pool. The calculated dose rate during transfer is 6 mrem/hr at the pool surface. When fuel is stored in the spent fuel storage pool racks, a minimum cover of 12 feet is provided.

6. What precautions are being taken to ensure that a rupture of a line penetrating the containment wall will not cause containment rupture?

Piping restraints will be provided at the containment penetrations to assure that failure of the pipe will not tear the containment. Drywell penetration nozzles will be "guarded" so that rupture of the pressure piping will not rupture the containment penetration nozzle.

7. State the manner in which the biological shield will be cooled and the design criteria to be followed in designing this cooling system.

The biological shield will be cooled in the vicinity adjacent to the core with treated cooling water from the reactor cooling water system flowing in embedded pipe coils. The system will be designed for a maximum concrete operating temperature of 150°F. Temperature gradients will be minimized to hold the temperature difference across the shield to 80°F or less. Should design calculations indicate high local gradients, consideration will be given to making the inner belt shield non-structural and insulating it from the outer structural concrete. Corrosion problems will be considered and some standby capacity will be available.

8. What are the closing time specifications on the containment isolation valves and turbine trip valves?

The turbine stop valves will close in less than 1 second. In the event of a main steam line rupture outside the drywell, the main steam line isolation valves will close in time to permit continued feedwater flow to recover the core before damaging temperatures are reached. This time has not been determined yet, but is expected to be within the range of 10 to 30 seconds. Solenoid valves used in pressure suppression system vent lines will close in about one second. Closing times for other isolation valves have not been determined.

9. State whether the drywell will be provided with an airlock entrance and indicate your intentions relative to entering the drywell during operation. State the type of instrumentation which will be provided to indicate recirculation pump performance. Indicate what assurance there is that the equipment will operate satisfactorily under 150°F ambient conditions.

   The dry well will be provided with an airlock entrance. It will not be entered during power operation, but may be entered when the reactor is hot and pressurized.

   Recirculation pump performance will be monitored by the following types of instrumentation. Alarm and read-out equipment will be located in the control room.

   Recirculation Flow - Flow metering nozzle in pump discharge piping.
   Electrical Power Input to Motor - Ammeter.
   Vibration - Vibration alarm switch on motor support.
   Mechanical Seal Performance - Pressures and temperatures at the individual seals are indicated, with alarm settings for selected high or low readings.
   Shaft Seal Leakage - A high flow reading is alarmed.
   Cooling Water Flow - Loss of cooling water flow is alarmed.
   Motor bearing Temperatures - High temperature is alarmed.
   Motor Winding Temperatures - High temperature is alarmed.
   Motor Lube Oil Levels - Selected high and low levels are alarmed.

   Recirculation pump operation under 150°F ambient conditions is expected to result in a winding temperature rise well within the capability of the Class B insulation supplied with the motors, projected over their expected life.

   Operation of the other pump components will not be affected by ambient conditions in the 150°F range.

10. Indicate how the spent fuel will be handled preparatory to shipment offsite.

   Tentative plans are to remove the fuel bundle channels, store the fuel bundles in the fuel pool for approximately 60 days or more, and then load them in an approved shipping cask for shipment to a reprocessing plant. Any damaged fuel bundles would be placed in water-tight containers for both storage and shipment. An alternate being considered is the removal of bundle structural components by disassembling the bundles within the fuel pool and then shipping the fuel rods to a reprocessing site and the structural components not suitable for reuse to an approved waste disposal site. No cutting or removal of fuel rod cladding is planned.

11. How many scrams can be accomplished with the stored energy in the scram accumulators.

One. The accumulators will be sized so that each drive can be completely scrammed with the water from its accumulator when the reactor is at atmospheric pressure. Recharging is necessary after each scram. The recharging action is automatically accomplished by the hydraulic system pumps after scram. Interlocks prevent drive withdrawal before recharging is completed.

12. State the core inlet subcooling at rated conditions.

13°F.

13. What is the reactivity worth of starting a recirculation pump under normal and abnormal conditions? Are interlocks provided to (a) prevent starting a pump when the loop valve is open and (b) limit the rate of valve opening?

Design of the recirculation system is not complete, but it will be designed so that the normal startup of a pump will not cause a significant reactor flux transient. As for abnormal conditions, analog computer studies have been made of the transient which would result if failure of the interlocks permitted the pump to start up with the discharge valve open. With the pump inertia that has been specified for the Bodega pumps, about four seconds are required for the pump to reach full speed under these conditions. Assuming an initial reactor power level of 80%, the maximum excess reactivity resulting from this transient is about 20 cents and the maximum fission power reached is about 125%, assuming no high flux scram occurs. Analysis of the "cold water accident", where it is assumed a loop is started when filled with cold water, has not been completed, but the interlocks and limited opening speed of the discharge valves, as discussed below, will prevent a severe transient in this case.

Interlocks will be provided to prevent starting of a recirculation pump if the main loop valve downstream from the pump is open or if the bypass valve around the main loop valve is closed. The rate of valve opening will be limited by the operating speed of the motor operators.

14. What conditions, such as low condenser vacuum, can prevent use of the main feedwater pump?

Any condition which interrupts or prevents operation of the turbine generator unit will similarly affect operation of the main feedwater pump. A number of abnormal conditions, which might lead to damage to the turbine generator if allowed to continue will cause the turbine stop valves to close. Among these are (1) Low condenser vacuum, (2) Low oil level, (3) Loss of generator field, (4) High reactor water level, (5) Low bearing oil pressure, (6) Overspeed, and (7) High bearing temperature. During turbine coast-down following the trip the main feedwater pump would continue to deliver water to the reactor until its discharge pressure dropped below reactor pressure.

Loss of auxiliary power would result in loss of the condensate pumps, and hence loss of feedwater pump suction.

15. What is the reactivity worth of the liquid poison system and what is its rate of reactivity addition? What is the worth of the system when the refueling pool is connected to the reactor vessel?

The reactivity worth of the liquid poison system based upon cold conditions and total injection will be at least 20% Δk assuming that the reactor water is at its normal level and the poison is evenly distributed throughout. With this concentration the rate of reactivity addition will be at least 0.5% Δk per minute. If the poison system is actuated at a time when the refueling pool is connected to the reactor, the reactivity worth will be about 5% Δk. This figure assumes that the gate between the fuel canal area immediately above the reactor vessel and the fuel storage pool is closed before any significant amount of poison has diffused into the pool and also that the poison is evenly distributed throughout the reactor vessel and canal water.

16. What is the heat removal capacity of one tube bundle of the emergency condenser?

Each bundle will have a heat removal capacity of 1.8% of rated reactor thermal output.

17. Describe the research and development program which has been or will be conducted to support the use of the internal steam separators and indicate the separation efficiency expected. Indicate the data which are presently available upon which the design will be based.

The Bodega reactor has the following approximate steam separation specifications and dimensions, based on present technology.

System operating pressure	1075
Maximum system carryunder	0.25% steam by weight in recirculation water
Maximum moisture in steam leaving axial units	6-10% moisture in steam by weight
Maximum moisture in steam leaving dryer	0.1% by weight
Number of axial units	102
Dimensions of axial units	55 inches long by 13.5 inches O.D.
Average quality entering axial unit	9.6% by weight
Average flow per axial unit	435,000 lbs/hr
Separator pressure drop	7 psi
Standpipe height	6 feet
Normal water level variation due to load changes	20 inches.

Internal steam separating systems have been developed intensively by the General Electric Atomic Power Equipment Department. Three separate test facilities have been used for this development: one uses high pressure steam-water, and two use air-water at atmospheric conditions. Steam-water has been used where carryover is important, such as tests of dryers and of axial units. Air-water has been used when flow observation is important, such as preliminary testing of primary units. Air-water has also been used for large scale, high flow system tests concerned mainly with two-phase flow in a closed plenum, flow through separators, and flow of the separated water which has left the separators.

The major effort has been the development of an axial separator which would handle a wide range of flows up to the required maximum with a relatively low pressure drop and do this over a wide range of water level. Also, a method has been developed for predicting the water level gradient which exists at the water discharge plenum for a group of separators operating in parallel.

Tests show that steam approaching the steam dryer after discharge from the axial separator must not exceed certain functions of velocity and inlet moisture. Dryer performance is also a function of operating pressure.

18. State the criteria to be followed in sizing the plenum space in each fuel rod and in preventing collapse from external pressure.

<u>Design Criteria to Avoid Collapse</u>. The clad thickness is established by determining the requirements to avoid collapse of the fuel rod. This is accomplished by (1) totalling the stresses related to external pressure, stress due to ovality of the tubing, and thermal stress; and (2) selecting the clad wall thickness such that the total stress does not exceed the yield strength of the material at any operating condition.

Criteria to Establish Plenum Space. The plenum space is established by totalling the effects of maximum internal pressure in any fuel rod due to gaseous fission products (at maximum anticipated UO_2 burnup) and trace volatile impurities in UO_2 pellets; and setting the plenum space to avoid a stress in excess of the yield strength of the cladding.

19. Describe the techniques and equipment to be used for decay heat removal during the first hour after shut down from full power under the assumption of failure of the entire feedwater system.

The emergency condenser plus the "bleed" valves in the "feed-bleed" system will handle decay heat removal in this case, even if one emergency condenser coil is unavailable. The bleed valves will automatically open if the emergency condenser alone will not handle the cooling load and will automatically close again as pressure is reduced before enough water is lost to approach uncovering the core. Backup for this method of cooling is available from the feed-bleed system. Feed can be supplied from the feed-bleed system pump or startup feed pump through core spray system piping which is still available in the event of a failure in the main feedwater system.

20. What is the heat removal capacity of the emergency core spray system? Assume a major rupture of a recirculation line has caused the water level to fall below the core.

Each of the two core spray system pumps is specified to deliver 1200 gpm at 150 psi. Heat removal capacity of the system depends on fuel cladding temperature. Assuming a fuel cladding temperature of 1500°F at the hottest spot, it is estimated that a 1200 gpm spray rate would give a heat removal capacity of at least 40 Mw.

Since submission of the Preliminary Hazards Summary Report, provisions for a source of high pressure core spray, in addition to the low pressure system described above, have been made. Piping will be installed so that flow from the start-up feed pump can be routed to the core spray header. Rated capacity for this pump is approximately 700 gpm at 1060 psi. With a cladding hot spot temperature of 1500°F, it is estimated that this flow should give a spray cooling capacity of at least 30 Mw.

21. What are the consequences of a leak in a recirculation loop that causes the water level in the core to fall in spite of maximum feedwater flow while the vessel pressure remains above 150 psi?

In this case the start-up feed pump flow would be routed through the core spray header to provide core cooling and reduce reactor pressure to below 150 psi at which time the low pressure core spray system would be started to continue core cooling.

22. Do any pipes or tubes, such as pneumatic or hydraulic lines, lead from the drywell area or the suppression pool tank to the control room? If so, discuss the potential for the transmission of radio-activity through these conduits to the control room if a severe accident occurs in the drywell.

No pneumatic or hydraulic lines will lead from the dry well area or the suppression pool tank to the control room. Instrument readings, such as pressure and level, will be transmitted to the control room electrically.

23. Show that the startup source will be capable of producing a satisfactory response in the nuclear instrumentation in view of the proposal that the nuclear instrumentation is to cover only a range of nine decades.

The design criteria for the strength and number of neutron sources is that the source or sources be of sufficient strength to provide one count per second on the startup instrumentation channels with a signal-to-noise ratio of 3 to 1. This may be difficult to achieve with the out-or-core startup channel detectors described in the Preliminary Hazards Summary Report.

Use of in-core detectors would reduce source requirements and give increased sensitivity. In-core startup detectors are being developed and will be used to replace the out-of-core detectors if they prove to offer significant advantages.

24. Justify the omission of the following as reactor scram signals:
(a) Seismic shock.
(b) Low control rod drive system hydraulic pressure.
(c) Low reactor pressure.

(a) The plant will be designed to ride safely through the maximum potential earthquake at the site. Even if it were not so designed, it is believed the addition of a seismic scram would not significantly add to plant safety because the consequences of the types of damage that would be expected from earthquake shocks would be essentially the same whether the reactor were scrammed on the initial shock or not. For this reason and to assure continuity of service to the system, we believe it essential to conservatively design the plant to withstand earthquake forces without seismic scram devices.

(b) It is not believed necessary to scram on low hydraulic system pressure because loss of system pressure would not result in loss of pressure in the scram accumulators. Additionally, a control rod drive can still be scrammed from reactor pressure if its accumulator loses pressure when the reactor is pressurized. Low accumulator pressure is alarmed and interlocks prevent control rod withdrawal if any two of the 145 accumulators lose pressure.

(c) A decrease in reactor pressure will cause an increase in core void content and a consequent decrease in reactor power due to the negative reactivity effect of the increase in voids. Past experience with other boiling water reactors and preliminary studies of the Bodega reactor indicate that the void shutdown mechanism is quite strong and that a low pressure scram is not necessary for reactor protection.

25. The reactor scram system includes a number of solenoid valves and other electrical devices. Is the reactor automatically shut down any time there is an interruption of electrical power to these devices or to the control chain which actuates these scram devices?

The power for the dual channel safety system, including the solenoid operated scram pilot valves, is supplied through two independent buses. Loss of power to both buses will cause the reactor to scram. Failure of power to only one bus will cause de-energization of the scram pilot valves connected to the channel served by that bus but will not result in a scram. This permits reactor operation to continue with single channel protection provided by the unaffected channel.

26. Explain how the two-channel reactor protection system is "fail-safe" if both channels must be de-energized to produce a reactor scram. Is there any way, such as transistor shorting, in which a channel may fail and yet maintain an energized output?

The two channel reactor protection system is "fail-safe" in that a number of component failures will not interfere with the reactor protection function. Failure of individual components in an unsafe manner is protected against by redundancy of both components and circuitry. Unfortunately the schematic diagram in the Preliminary Hazards Report does not show this, but the component and circuit arrangement is such that it would require a minimum of four particular transistors (one of each of four pairs) or two series pairs of diodes (one pair in each of two sensing circuits) failing simultaneously to disable one safety channel in a manner which would prevent reactor scram. Failure of components is detected by periodic tests and the gross failure of the number of components required to disable a channel is believed to be highly improbable. A new schematic drawing showing one of the two channels is shown on the next page.

GENERAL ELECTRIC

SK56100-528

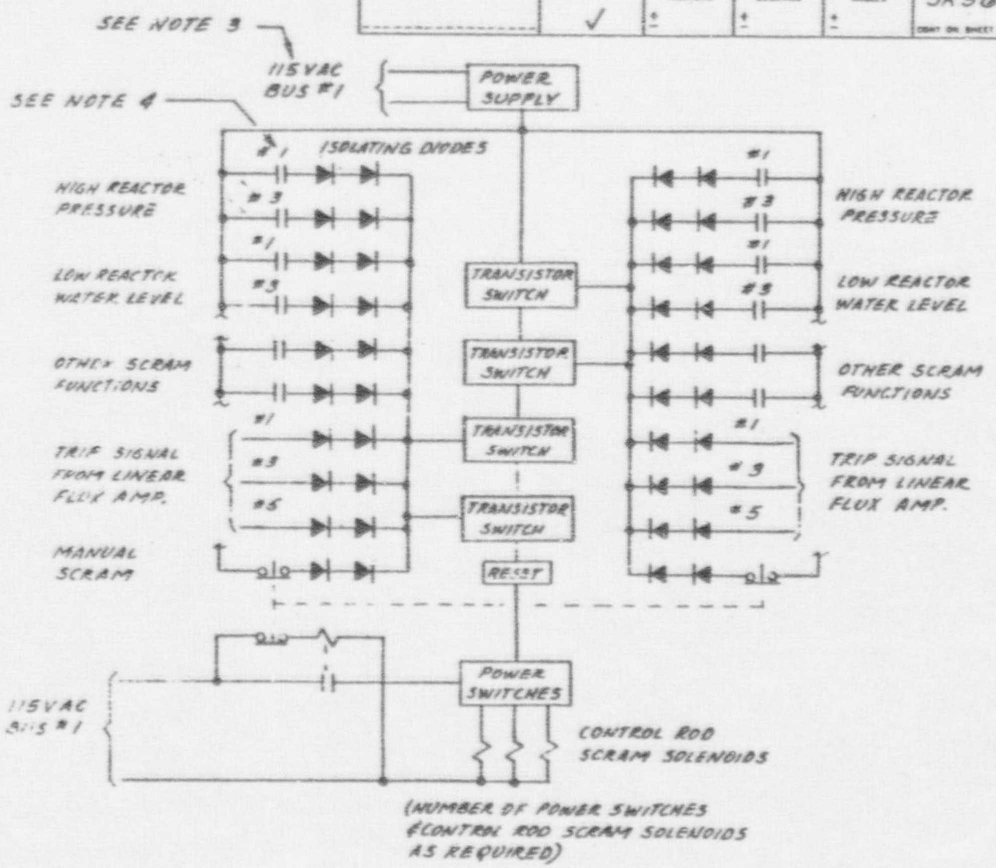
UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:—

APPLIED PRACTICES	SURFACES	TOLERANCES ON MACHINED DIMENSIONS: FRACTIONS	DECIMALS	ANGLES
	✓	±	±	±

REV. NO. 0 — SK56100-528 — CONT ON SHEET FL — SH NO. 1

TITLE: REACTOR PROTECTION SYSTEM

FIRST MADE FOR BODEGA BAY



CHANNEL 1

SEE NOTE 1, 2

NOTES:

1. CHANNEL 1 SHOWN AT LEFT IS TYPICAL FOR CHANNEL 2 (NOT SHOWN) EXCEPT AS NOTED.
2. ALL SCRAM SIGNALS RECORDED ON OPERATIONAL RECORDER.
3. CHANNEL 2 EXCITED FROM 115VAC BUS #2.
4. CHANNEL 2 SIGNAL CONTACTS ARE EVEN NUMBERS. (2, 4, 6 ETC.)

REVISIONS

PRINTS TO

MADE BY MERTEN MAR. 4, '63 — ISSUED — APPROVALS

APED — SAN JOSE

SK56100-528 — CONT ON SHEET FL — SH NO. 1

SK56100-528 FL 1

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27. Describe the shielding that will be placed below the reactor vessel to permit maintenance operations in the "sub-pile" room.

    The 14 feet of water between the bottom of the core and the reactor vessel bottom head provides adequate shielding for maintenance of the control rod drives when shut down. No additional shielding is contemplated at this time.

28. If an earthquake or some other occurrence should interrupt the supply of sea water, what auxiliary sources of cooling water will always be available and how long could these water supplies be used to dissipate reactor decay heat?

    It is not expected that the sea water cooling system would be interrupted by an earthquake. In addition, the raw water storage system will have a minimum reserve for fire and other emergencies of 150,000 gallons of fresh water. This amount of water could supply makeup to the emergency condenser for approximately 2 days after power shutdown. After that time other emergency sources such as fire pumps could be available if needed.

29. What reactor operations resulting in reactivity changes will be permitted while the drywell vessel head is removed?

    Core loading and refueling operations including poison curtain movement, rod withdrawals to verify that shutdown margin requirements are being met, and critical testing at power levels up to about 0.1 Mw.

30. What water clean-up equipment and off-gas removal equipment will be provided for the removal and control of fission products in the fuel storage pool?

    The water in the spent fuel storage pool will be recirculated through a filter to control cleanliness and visibility. Should fission products be released to the pool water in quantities or form which the filter can not handle, the water can be routed through the radwaste demineralizer. The surface of the pool will be swept with the building ventilation air and exhausted out the stack.

31. Explain the criteria to be followed in determining whether both or only one of a pair of isolation valves is to receive automatic closure signals.

One isolation valve is provided at the containment barrier for each major penetration, as described in Section III-2-f of Exhibit C. Each such isolation valve is backed up by a remotely operable process valve or by a second isolation valve (in some instances, a check valve). Where two isolation valves are provided, both will receive automatic closure signals.

32. What is the magnitude of the pressure rise in the Refueling Containment Building in the event of a failure of a reactor auxiliary system (for example, a leak in the emergency condenser system or the high pressure demineralizer)? What is the design pressure rating of the Refueling Containment Building?

The pressure rise in the refueling building which would result from a failure of a reactor auxiliary system is dependent on such things as the size of the break, isolation valve timing, and the extent of guard piping, which have not been completely determined. Should the calculated pressure rise be such as to endanger the building structure, pressure relief will be incorporated into the design, using non-structural relief panels or doors.

The earthquake resistant design of the building affords approximately 12 inches of water of pressure rating.

33. Justify the use of the proposed minimum burnout ratio criterion for the design of the core. What burnout data correlation has been used in burnout analyses? At what reactor power level is the burnout ratio of 1.5 expected to occur? At what reactor power level (steady state) would the burnout ratio reach 1.0? Discuss the nature of any conservatism used in estimating the heat fluxes used in burnout ratio calculations. What minimum burnout ratio is reached as a consequence of a loss of feedwater flow accident?

The burnout correlation being used for design and analysis of the Bodega Bay plant is given in "Burnout Limit Curves for Boiling Water Reactors," by E. Janssen and S. Levy, April 14, 1962 (APED 3892). This correlation is also used for burnout calculations at the Dresden and Big Rock Point Nuclear Power Stations. The minimum burnout ratio of 1.5 proposed for Bodega Bay is the same as that specified in the Big Rock operating license.

In discussions held with ACRS and AEC staff members, preliminary to issue of the Big Rock operating license, the question of the minimum burnout ratio was thoroughly reviewed. Several important points made in the discussions are summarized as follows:

1. Emphasis on experimental determination of burnout in recent years has resulted in a considerably increased number of data points and more accuracy in experimental techniques as indicated by a decrease in the scatter of experimental data.

2. Investigation of variations in the geometries of test assemblies has verified the applicability of experimental results from simplified geometry tests to multi-rod configurations.

3. Better understanding of the physical phenomena involved in burnout has been attained through actual observation and the development of consistent theoretical models.

4. Preliminary investigation of operation beyond "burnout" (departure from nucleate boiling) has been carried out which indicates that such operation can be sustained without actual burnout of fuel.

5. The Vallecitos Boiling Water Reactor has been operated over a year with a minimum burnout ratio of 1.5.

The factor of 1.5 is applied to the basic burnout correlation to account for any uncertainties associated with the experimental data and development of the correlation. As a result of the work summarized above, these uncertainties have been considerably reduced since reactors such as Dresden were designed with a minimum burnout ratio of 2.0. Based on the use of this improved burnout correlation, a minimum burnout ratio of 1.5 is considered justified for use in designing the Bodega Bay reactor core.

Because we evaluate burnout margin on the basis of abnormal localized peaking, rather than overall power margin, we do not have detailed calculations in answer to this question. We estimate that if core power were increased, maintaining a fixed power distribution, the reactor power correspoinding to a 1.5 burnout ratio would be in excess of 120% of rated power. We are not able to extrapolate to a burnout ratio of one without further detailed calculations that are not now completed.

The conservatism used in estimating the heat fluxes used in burnout ratio calculations is as follows:

a. The heat fluxes are calculated for core average and local overpower conditions which have been conservatively estimated.

b. Less than 1% of the fuel in the core would be operating at the peak calculated heat flux even if all gross and local peaking factors were to occur at the same location in the core.

A sudden loss of feedwater flow while the reactor is at rated power will cause the level to drop at a rate of 4 inches per second. The low level scram point will be set at about 12 inches below the normal water level. Thus, low level scram will occur about 3 seconds after loss of feedwater flow. Since the transit time from the feedwater sparger ring to the core inlet is about 10 seconds, the reactor will have scrammed before the core inlet temperature can rise appreciably and then the burnout ratio will not decrease during this transient.

For slow loss of feed flow the core inlet temperature can rise appreciably prior to low level scram. Preliminary studies indicate that reactor power will drop slowly due to the increased core voids caused by the increase in core inlet temperature. In this case the reactor pressure regulator will decrease steam flow in order to hold pressure constant. The slight increase in core voids necessary to cause reactor power to decrease will not cause any appreciable change in the burnout ratio since, essentially, steady state conditions will be maintained throughout the slow loss of feed flow.

If the question refers to loss of reactor recirculation flow, it has been calculated that the minimum burnout ratio will not fall below 1.5 in the event recirculation flow was reduced due to loss of pump power. The recirculation pumps are equipped with flywheels to minimize the reduction in burnout ratio during the transient.

34. State where poison curtains will be located in the core, how they will be supported, and how they will be held in place.

The poison curtains will be located in the gaps between fuel assemblies that do not contain movable control blades. The curtains will be supported from a hanger rod which attaches to the top guide structure. They are held in place by the fuel element channels and significant upward or downward movement is prevented by the design of the hanger rod attachment to the top guide structure. Should the hanger fail, movement out of the core is prevented by the top guide structure and the fuel support plates.

35. State the average steam volume fraction in the core at rated conditions and the average exit as well as the hot channel exit volume fractions. Describe the research and development and/or analytical programs which have been or will be conducted to determine the stability of the reactor at proposed conditions.

Preliminary calculations indicate that at rated conditions, the steam volume fractions are as follows:

core average - 37%
channel exit average - 58%
hot channel exit - 67%

In addition to development work being carried out by General Electric Company in the general area of two-phase flow dynamics, there has been considerable effort within the Bodega project to assure the stability of the reactor. Specifically, this effort includes the development of an analog computer model for plant dynamic studies. This model has

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been checked against Dresden reactor plant transients, both for Dresden rated conditions and for the conditions of the Dresden High Voids Test (30% core average voids) conducted in September, 1961. The model has been found to give adequate correspondence to test results. This model provides for calculation for transient conditions of reactor power, pressure, recirculation flow, core voids, fuel temperature, water level, steam flow, feed flow, and subcooling in such a manner as to adequately describe the interaction of these variables for purposes of evaluating plant dynamic behavior. The dynamic characteristics of the internal steam separators are important in developing the analog model of the Bodega plant. The internal separators are the only area of the Bodega model which is substantially different from the Dresden system. In conjunction with steam separator development tests, tests are planned to experimentally confirm dynamic characteristics now used in the Bodega model. It is believed that parametric studies made with the Bodega model will allow the plant to be designed in such a manner that its satisfactory dynamic performance will be assured.

36. State the design shutdown margin for the most reactive condition at the most reactive time in core life.

The criterion for design is that the core be subcritical by 1% Δ k with the most reactive control rod fully withdrawn at the most reactive condition at any time in core life. With all control rods inserted, the shutdown margin will be at least 3% Δ k.

37. Describe the catch basins and how they will function. Discuss whether they can overflow. Estimate maximum radioactivity concentrations to be allowed to accumulate in a catch basin, both in solution or suspension and as deposited material.

Normally catch basins in the plant areas subject to possible radioactive contamination continually drain rain water to the circulating water tunnel and thence to the ocean. The catch basins can be isolated from the tunnel and pumped or drained to Radwaste for treatment or storage. This will be done when equipment cleanup work is being done, or when radioactive material is in the drainable areas.

These catch basins will be designed to handle, without overflowing, an estimated peak runoff due to rainfall with discharge to the ocean. They will not overflow during cleanup operations of either equipment or equipment areas, and no radioactivity concentrations will be permitted to accumulate in the catch basins but will be drained off to Radwaste for treatment or storage.

38. Provide an analysis, to demonstrate that the proposed design, operating procedures, and release limits for the circulating water and waste disposal systems in conjunction with the proposed monitoring equipment and procedures assure that no person will be exposed to concentrations of radioactivity in excess of those permitted by Commission regulations if the marine environment is subjected to the effects of the operations contemplated. This analysis should take into account re-concentration effects and other pertinent factors.

Our replies to question 1 indicate the scope of research programs being carried out at the Bodega plant site on meteorology, oceanography, marine biology and a pre-operational radiological survey. Data and information gathered from these research programs will provide the basis for establishing release limits for the circulating water and gaseous waste disposal systems. The oceanographic surveys will indicate the capacity of the ocean to diffuse and mix the condenser cooling water. Along with such physical field tests analytical methods have been pursued which indicate that based on the expected radionuclides and their expected concentrations a general control limit of 10^{-7} microcuries per cc, or separate limit based on batch analysis, will provide ample safety with regard to hazard to man from seafood. Biological reconcentration factors as suggested in appropriate NAS documents were included in this evaluation. Diffusion in the sea using several conservative analytical models indicates a factor of 200 to 1000 within one kilometer of the discharge point. Additional analyses are being made by the Company's marine consultant, Dr. Salo of Humboldt State College, and his staff. Because of the nature of the outfall and the rugged coastline, access to the outfall proper by either fish or fishermen is practically precluded.

Gaseous release limits are dependent on local meteorological conditions combined with local topography, stack height and quantity of gaseous waste to be discharged. After sufficient data is collected an analysis will be made to demonstrate that the proposed design, operating procedures and release limits for gaseous wastes will be such that no person will be exposed to concentrations of radioactivity in excess of those permitted by Commission regulations.

In addition, an adequate environmental monitoring program will be conducted to assure that permissible exposure rates due to plant waste emissions will not be exceeded.

39. What is expected to be the total annual discharge of radioactive liquid and gaseous wastes from this plant?

We plan to request liquid and gaseous waste disposal limits based on applicable limits in 10 CFR 20.

It is expected that normal operation discharges will be well within expected permissible limits but we are unable to estimate how much lower they will be until additional operating experience is gained at Humboldt Bay and other reactors similar in design to Bodega.

40. Delineate the conditions under which (1) the emergency condenser will be in service while other containment isolation valves are closed, (2) the emergency condenser isolation valves will be closed while other containment isolation valves are open and (3) all isolation valves are closed.

(1) The emergency condenser is automatically brought into service by an indication of reactor high pressure. Reactor high pressure could result from closure of the main steam isolation valves during power operation. Automatic closure of the main steam isolation valves is initiated by the following conditions:

(a) Low-low reactor water level
(b) Main steam line leak in the pipe tunnel
(c) Low vacuum in the main condenser

(2) The emergency condenser isolation valves close automatically on indication of an emergency condenser piping leak. (The method of indicationhas not yet been selected.)

(3) A situation involving an emergency condenser piping leak coincident with conditions (a),(b), or (c) above has not been specifically provided for in the design of this unit. However, if such a coincidence were to occur, the isolation feed pump could supply cooling water to the reactor by way of the core spray line. The core spray system would be a backup for the isolation feed and bleed system.

41. What are the consequences of an accident in which the feedwater system fails and operator action or a reactor high pressure signal causes the isolation bleed valves to open?

The bleed valves will automatically close again when reactor pressure drops below 1150 psi and decay heat removal will be handled by the emergency condenser, which comes into operation at 1100 psi reactor pressure. The reactor would have scrammed from a low water level or other signal. The water level over the core is sufficient to dissipate decay heat for 15 minutes after scram, but actually only a fraction of this would be used even if only one emergency condenser coil were available. If the condenser were completely unavailable, the startup feed pump would pump water through the core spray lines to keep the core covered.

42. Describe the gas holdup system and state the system capacity.

The off-gas holdup system for the air ejector discharge comprises a low velocity pipe line between air ejector and stack isolation valve, to provide for decay of short half-life radioactive gases, and to permit time for operator action before release. Gases leaving the pipe pass through an absolute filter and off-gas isolation line before being released to the stack. The off-gas monitor system draws a sample from ahead of the holdup pipe, measuring the activity and providing a signal to close the off-gas line isolation valve before maximum permissible release rates are exceeded. Other components of the system include drains, combustion suppression, purging and recirculation features and an isolation valve at the beginning of the holdup pipe. The holdup pipe will have an approximate internal volume of 2300 cubic feet.

The gland seal exhauster has a holdup pipe of approximately 1400 cubic feet volume.

43. How far from the nearest earthquake fault or branch fault is the reactor to be located? How far from the San Andreas fault is the reactor to be located?

The geologic and seismologic characteristics of Bodega Head have been carefully investigated by the Company's consultants, Mr. William Quaide, geologist, and Mr. Don Tocher, consulting seismologist, of University of California at Berkeley. In addition, extensive soil borings were conducted to determine soil and rock conditions on Bodega Head and to determine whether or not rock faulting exists in the selected power plant site. It is the conclusion of the Company's consultants and verified by the borings that no active faulting exists on Bodega Head and particularly under the power plant site. The geologic map in Appendix IV, which was prepared by Mr. William Quaide indicates the western margin of the San Andreas fault zone. The distance from this western margin to the reactor is approximately 1,000 feet. The fault zone at this point is estimated to be about a mile and a half wide. Since there are no active branch faults on Bodega Head the western edge of the San Andreas fault is therefore the closest known active fault line to the plant site.

44. Describe the research and development program which will be conducted to support the belief that a 2700 Mw-sec. nuclear excursion will not damage the reactor pressure vessel. Indicate any design features which could reduce the likelihood or magnitude of a rod dropout accident.

Development of Rod-Dropout-Accident Analytical Models - Work is continuing on the development of improved mathematical models to predict the consequences of high reactivity addition rate accidents such as the rod dropout accident. These models are checked against experimental data obtained from transient tests conducted in the AEC's Spert facilities with uranium dioxide fuel and thus the work is dependent upon the Spert program in this report. Present models accurately predict the consequences of accidents up to the threshold of fuel damage, which is as far as the Spert tests have gone to date. Additional data on transients large enough to damage fuel are needed to verify the models used to predict the consequences of more severe accidents. It is understood that tests past the fuel damage threshold are planned at Spert within the coming year. It is believed that the analytical development efforts can be completed following the Spert tests. It should then be possible to predict the consequences of high reactivity addition rate accidents and to evaluate the need for rod worth minimizer computer and/or rod dropout velocity limiter devices described below.

DESIGN FEATURES - Design features which reduce the likelihood or magnitude of a rod dropout accident are as follows:

1. Use of a large number (145) of control rods. This makes it possible to plan operating rod patterns so that the dropout of a rod of normal reactivity worth would not result in fuel melting and the dropout of the highest worth rod from a normal pattern would not result in fuel rod rupture as a result of fuel vaporization.

2. Design of the control rod blades with ample clearances and other features to minimize the potential for the blade to stick within the core. Experience with blades of similar design indicates that they are not subject to significant dimensional distortions of any kind as a result of exposure to a reactor environment.

3. Use of a minimum-clearance, high-strength coupling between the control rod blade and its drive which minimizes the possibility of a separation of a blade from its drive.

4. The drives will be designed to allow a check of coupling integrity to be made on withdrawn blades.

5. Use of sensitive nuclear instrumentation that will make it possible to verify the control blade follows the drive during major movements when the reactor is at or near critical.

6. In addition to the above features, a rod worth minimizer computer and a rod dropout velocity limiter device are being developed for possible use in the Bodega plant. These are described below:

   Rod-Worth-Minimizer Computer - A computer which would continually monitor control rod patterns to reinforce procedural controls used to insure that patterns giving undesirably high reactivity worths for individual rods are not set up is being developed and will be used in the Bodega plant if successfully developed.

   Rod Dropout Velocity Limiter - Conceptual designs of flow restricting devices to be attached to the control rod blades to

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limit potential dropout velocities to safe values in the unlikely event that a blade becomes stuck, separates from its drive, and then works loose and falls after the drive is retracted. The device must, of course, be designed in such a way that it does not significantly decrease scram insertion rates. Several designs are now under study and the most promising of these will be subjected to development testing and, if successfully developed, may be used in the Bodega plant as an alternate or in addition to the rod worth minimizer computer described above.

45. Discuss the possible effects on this plant should an earthquake cause displacements along minor faults under the plant.

The plant will be designed to resist seismic forces corresponding to Richter magnitude 8.2 (established by Dr. G. W. Housner) which is equal to the severest shocks recorded or estimated in California. As indicated in the reply to question 43, the geologic and seismologic characteristics of Bodega Head were carefully investigated by the Company's consultants Mr. William Quaide, geologist, and Mr. Don Tocher, consulting seismologist of the University of California at Berkeley. In addition, extensive borings were conducted to determine soil and rock conditions on Bodega Head and to determine whether or not rock faulting existed in the selected power plant site. Mr. Quaide and Mr. Tocher conclude that there is no evidence of any active faulting on Bodega Head. They also conclude that there is no evidence of any active faulting having occurred at the plant site during the past several thousand years.

46. What are the consequences of a major steam line break in the pipe tunnel if the accident occurs during inversion conditions? Treat both a case in which the containment isolation valve functions properly and a case in which the isolation valve fails to close.

The analysis presented in the PHSR of a main steam leak in the pipe tunnel during unstable meteorology did not take credit for diffusion of the steam cloud by atmospheric turbulence. Only the initial expansion of the cloud was considered. Therefore, the same analysis would apply to an inversion condition if cloud rise is neglected and if the same 10 mph wind speed is assumed. If a different wind speed were assumed for an inversion, the calculated dose rate would be inversely proportional to the assumed wind speed.

If credit were taken for cloud rise in the inversion case, maximum ground concentrations of released halogens would be less by a considerable factor. It is expected that the cloud rise during an inversion would be very significant because atmospheric mixing would be poor and the steam cloud would remain as buoyant as air heated to approximately 600°F. Figure 65 of <u>Meteorology and Atomic Energy</u> indicates rises of thousands of feet for clouds equivalent to 600°F air.

A steam line break in the tunnel coincident with failure of the isolation valves to close has not specifically been provided for in the design because it has been considered incredible, in view of the frequent testing of the valves that will be carried out to assure they will function properly. However, if such an event were to occur, essentially all the reactor water could be discharged to the pipe tunnel compared to only 43% if the valve did operate. The physical dimensions of the steam cloud would be increased only 10 or 20%. Consequently, total exposure due to halogens in the passing cloud would be increased by only 10 or 20%, on the order of 2 or 3 rems at the nearest site boundary.

The core would be cooled down by the rapid flow of escaping coolant during the blowdown and core cooling would continue to be accomplished by the core spray system after the blowdown was completed. Continued feed flow would recover the core in approximately three minutes. (With reactor pressure at atmospheric, the condensate pumps are capable of pumping rated feed flow through the turbine driven feed pump, even if the feed pump has stopped.) Consequently, even if this incredible accident were to occur, the integrity of the fuel cladding would be maintained.

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

The Bodega MCOA calculations assumed time-rated removal of halogens and solids as experimental data reported in the literature indicate this to be more realistic than assuming an arbitrary fixed removal fraction. Preliminary calculations indicate that the integrated release of halogens would be less than a factor of ten greater if only a 50% credit were taken for plate out, as suggested in TID 14844.

48. Submit samples of the calculations used in determining the doses rates set forth in the Radiological Effects of Major Accidents.

The solution of Suttons equation for an unstable 10 mph wind at 0.6 miles assuming a 300 ft plume height is as follows:

($n = .22$, $Cy = .6$, $C_z = .2$, 10 mph = 4.47 m/s,

0.6 miles = 1000 m, 300 ft = 91.5 m)

$$Q/cc = \frac{2\,Q/sec \cdot 10^{-6}}{\pi C_y C_z u x^{2-n}} \exp\left(\frac{-1}{x^{2-n}} \cdot \frac{h^2}{C_z^2}\right)$$

$$\frac{Q/cc}{Q/sec.} = \frac{2 \times 10^{-6}}{\pi \cdot .6 \cdot .2 \times 4.47 \times 1000^{1.78}} \exp\left(\frac{-1}{1000^{1.78}} \frac{91.5^2}{.04}\right)$$

where Q/sec = stack discharge rate, quantity per second
Q/cc = ground concentration, quantity per cubic centimeter
Cy, Cz = diffusion coefficients, meters to the n/2 power
n = stability parameter, dimensionless
u = wind speed, meters per second
x = distance downwind, meters
h = height of plume centerline above ground, meters

For the stable case a moderately stable condition was assumed because the assumption of great stability results in insignificant concentrations reaching the ground out to ten miles or more. Parameters for great stability are given in Table 14 of Appendix III. For moderate stability the assumed parameters were $n = .3$, $Cy = .21$, and $Cz = .07$. This gives a maximum value of (Q/cc) per (Q/sec) of 9.4×10^{-12} at 1.75 miles falling off to 7.0×10^{-12} at 3 miles for a 5 mph wind speed and 200 ft plume center height. The value 9.4×10^{-12} was conservatively used for the estimates representing the hills on the main land 3 miles away. Consequently, the doses and dose rates for the moderately stable condition are $9.4/2.08 = 4.5$ times those for the unstable case.

The maximum stack discharge rate shown on Figure VIII-5 of the PHSR is 0.7 curies of noble gases per second for the Maximum Credible Operating Accident. From the above solution of Suttons equation this maximum calculated stack discharge rate would give a maximum ground concentration as follows:

$Q/cc = 0.7 \times 2.08 \times 10^{-12} = 1.4 \times 10^{-12}$ curies/cc

This maximum stack discharge rate would occur approximately one day after the accident. Percentage composition of various isotopes in fission product mixtures as a function of irradiation time and decay time can be obtained from HW-33414 "Computed Fission Product Decay". An infinite cloud containing 10^{-9} curies/cc of an equilibrium mixture (500 day irradiation) of noble gases after 24 hour decays would give a dose rate of approximately 0.2 rem/hr. A concentration of 1.4×10^{-12} curies/cc would give $0.2 \times 1.4 \times 10^{-3} = .28 \times 10^{-3}$ rem/hr for the MCOA.

If the design basis refueling accident were to occur 24 hours after shutdown, the same mixture and conversion factor would apply. The maximum stack discharge rate of 5 c/sec of noble gases would result in a calculated maximum ground concentration of $5 \times 2.08 \times 10^{-12} = 10.4 \times 10^{-12}$ curies/cc and a maximum dose rate of $10.4 \times .2 \times 10^{-3} = .002$ rem/hr. The corresponding dose rate for the moderately stable case would be $4.5 \times .002 = .01$ rem/hr.

The maximum discharge rate for halogens shown on Figure VII-5 is 2×10^{-3} curies/sec for the MCOA at 2 hours. From the above solution of Suttons equation this would result in a maximum ground concentration of halogens of $2 \times 10^{-3} \times 2.08 \times 10^{-12} = 4.16 \times 10^{-15}$ curies/cc. Inhalation of 10^{-12} curies/cc of a 2 hour old equilibrium mixture of halogens would give a thyroid dose rate of approximately .26 rem/hr. The calculated maximum thyroid dose rate for the MCOA would be $.26 \times 4.16 \times 10^{-3} = .0011$ rem/hr. The corresponding dose rate for the moderately stable case would be $4.5 \times .0011 = .005$ rem/hr.

Percentage composition of isotopes in mixtures of fission products varies with time. This was taken into account in integrating the exposure rates.

49. Desc:‚be the test program to be used to demonstrate the acceptability of the proposed control rod drive system and indicate the acceptability criteria.

The program consists of tests on prototype mechanisms in our San Jose Test Facility which include functional tests, calibration tests, overload tests, and endurance tests, all run under conditions of pressure and tempurature expected in the reactor. In addition, each drive intended for use in the reactor must pass functional tests which include leakage, friction, scram time, and shim operation before installation in the reactor. After installation, the shim and scram characteristic of each drive is determined periodically in conjunction with the reactor hydraulic system. Detailed procedures for these test programs and acceptability criteria have not yet been worked out but will be similar to those for the Humboldt Bay reactor drives which were described in Amendment 18 to the Humboldt license application.