Knuth

HAZARDS ANALYSIS

by the

DIVISION OF LICENSING AND REGULATION

in the matter of

PACIFIC GAS AND ELECTRIC COMPANY

BODEGA BAY ATOMIC PARK

UNIT NUMBER 1

CONSTRUCTION PERMIT

DOCKET 50-205

TABLE OF CONTENTS

May 27, 1963

APPENDICES

A. Geology & Seismology (Design Criteria)

B. Effects of Routine Discharge of Radioactive Materials to the Marine Environment

May 27, 1963

I. Introduction

The Pacific Gas & Electric Company (PG&E) proposes to construct and operate a nuclear power plant on Bodega Head in Sonoma County, California. According to the proposal PG&E will design and supervise construction of the unit, while the ~~Gen~~ General Electric Company will furnish the nuclear steam supply system and the turbine generator.

The proposed plant, designated by PG&E as Bodega Bay Atomic Park Unit Number 1, will be a direct cycle, forced-circulation boiling water reactor producing power ~~nuclear energy~~ at the rate of 1,008 megawatts (Mw). The gross electrical generating capacity will be approximately 325~~,000~~ Mw.

The Bodega Plant is similar in many respects to boiling water power reactors now in operation. Features of the plant which require research and development programs ~~or developmental effort~~ in order to provide engineering information necessary ~~in~~ for the design or evaluation of the nuclear plant will be discussed in Section IV of this analysis.

II. Background

On ____________ PG&E submitted an application to the AEC for a construction permit and operating license pursuant to Title 10, Chapter 1, Code of Federal Regulations, Part 50 (10 CFR 50). The application, which includes a "Preliminary Hazards Summary Report", dated December 28, 1962, and Amendments 1 and 2 to the application dated March 4 and April 5, 1963, respectively, ~~has~~ have been reviewed

May 27, 1963

by the staff of the Division of Licensing & Regulation in accordance with 10 CFR 50. The application has also been considered by the AEC's Advisory Committee on Reactor Safeguards (ACRS), as required by the Atomic Energy Act and the regulations of the AEC. The views of the ACRS, as expressed in a letter of April 18, 1963, (a copy of which is attached hereto as Appendix _____) were also considered in the regulatory staff's evaluation.

At this time there are a number of features of plant design and operation which have not been definitely resolved. The Commission's regulations provide for the issuance of a construction permit on a provisional basis ~~provisional basis~~ in cases, like ~~as~~ this, where there are ~~in which~~ some aspects of design which have not been completed. A provisional construction permit may be issued, according to Section 50.35, 10 CFR on the basis of findings, among others, that (1) the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components on which further technical information is required; (2) the omitted technical information will be supplied; (3) the applicant has proposed, and there will be conducted, a research and development program reasonably designed to resolve the safety questions, if any, with respect to those features or components which require research and development; and that (4) on the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility and (ii) taking into consideration the site criteria contained in Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

The permit, if granted, would authorize construction only. The Commission would require ~~an~~ timely reports from PG&E with respect to results of research and development and final design of the more significant design features. The AEC staff would continue its evaluation of the safety of the plant in light of this information. An operating license will not be issued until the final design has been evaluated by the AEC staff and the ACRS. In addition, the definite plans for operations would be evaluated by these two groups.

Pursuant to a Notice of Hearing published ________________, the issuance of a provisional construction permit to PG&E will be considered at a public hearing to be held ________________, at ____________________a.m., PDT, on ____________________1963 before an Atomic Licensing and Safety Board appointed by the AEC. The issues to be considered at the hearing are:

(To be provided by OGC)

(Statement of position of staff at hearing)

The staff's evaluation of the proposed Bodega nuclear power plant and its position on the issues at the forthcoming hearing are based on all the technical information submitted as part of the applicant's request for a construction permit and the report from the ACRS. All of this information is available for inspection and review at the Commission's Public Document Room in Washington, D. C., and at the Commission's San Francisco Operations Office, ______________________________, San Francisco, California. This evaluation and proposed recommendation is subject to modification in the light of any further information which may become available, including the evidence introduced at the hearing. Under the Commission's regulations, any person whose interest may be affected may file a petition to intervene and, if granted, may participate in the proceeding. The decision of the Commission will be based upon the entire record in the proceeding.

May 27, 1963

III. Plant Description ~~and Safety Analysis~~

The Bodega Reactor is a direct cycle, forced circulation boiling water reactor with internal steam separation. Nuclear energy released in the reactor at the rate of 1,008 megawatts will be transferred to the ~~boiling~~ water coolant which is circulated ~~circulating~~ through the reactor. ~~Fuel elements with~~ The steam which is generated at 1,075 psia within the reactor flows to a turbine generator; ~~with a~~ the gross electrical generating capacity of the generator is ~~of~~ about 325 megawatts. Reactor ~~cooling water~~ coolant is recirculated through four recirculation loops, each ~~containing~~ loop contains a pump rated at 29,000 gpm. After passing through the turbine, the steam will be condensed and then passed through a demineralizer. ~~and the condensate after demineralization will be~~ The demineralized condensate will be returned to the reactor vessel. The reactor coolant ~~water,~~ which ~~will~~ contain ~~some~~ radioactive materials, will be ~~circulated~~ contained within a closed system from which the only normal effluent will be the continuous discharge of noncondensible gases. This gaseous material which will contain radioactive gases will be monitored and released from the reactor stack. As in conventional power plants, the condenser will be cooled by water drawn from nearby sources. In this case water to cool the condenser will be taken from Bodega Bay and discharged into the Pacific Ocean. ~~from time to time~~ Regulated quantities ~~from~~ of radioactive liquids will be discharged to the ocean via ~~in~~ the condenser coolant water.

An overall judgment concerning safety of reactor operations or the acceptability of potential hazards must be based upon a number of individual safety considerations. In ~~the final analysis many of these considerations~~

May 27, 1963

~~In the final analysis many of these~~

In the final analysis many of these considerations require the detailed study and evaluation of design features and operations which are not ~~cannot be~~ clearly defined at this time. In this case, PG&E has presented a hazards report which contains general descriptive material concerning the conceptual design of the proposed plant ~~plans~~, known facts concerning the proposed site and its environment, and an analysis of the radiological effects of normal operation and accidents on the surrounding area. The present evaluation by the Commission staff is therefore based upon the principles of design rather than upon details of design themselfes. In the case of features which are of particular importance to safety, the staff expects to receive and intends to require information on final design of these features before PG&E has expended any substantial amount of effort in the construction of those features.

In the following sections of this report the staff discusses the more important safety considerations which have led to its conclusions with respect to operation~~s~~ of the proposed plant without undue risk to the health and safety of the public.

May 27, 1963

III. A. Site and Environmental Factors

(by John Newell)

May 27, 1963

III. B. Special Earthquake Design Considerations

(To be written after complete study of consultants' reports)

May 27, 1963

C. Containment

The containment system proposed for this facility is one which ~~depends upon~~ utilizes the pressure suppression concept. ~~Its~~ The ~~xxxifx~~ design is similar in many respects to that used at Humboldt Bay. Significant features of the Bodega Bay Plant design include the following:

1. Plans for the Bodega Bay Plant call for a dry well having a 60 ft. diameter spherical lower section and a 26 ft. diameter cylindrical upper section. The overall height ~~[illegible]~~ of the dry well is approximately 100 feet.
2. Four reactor recirculation loops, ~~each with a pump, and~~ The reactor vessel with the ~~the reactor vessel will~~ ~~make~~ will be located within the dry well.
3. The dry well will have an airlock entrance. Personnel entry is not planned during reactor operation, but is contemplated with the reactor hot and pressurized.
4. The suppression chamber will be in the form of a torus and will have a major diameter of 93 ft. and a cross section diameter of 26 ft.

Both the dry well and the suppression chamber will be designed and constructed in accordance with the ASME boiler and Pressure Vessel Code, Section VIII. Piping restraints will be provided at containment penetrations to assure that failure of the pipe will not cause containment rupture. A concrete refueling building will contain the dry well and suppression chamber. Pressure and leak rate specifications for these containment system components are as follows:

Component	Design Pressure	Leak Rate (% of volume in 24 hours)
Dry well	62 psig	0.5 (at design pressure)
Suppression chamber	35 psig	0.5 (at design pressure)
Refueling building	12 in. H_2O	100 (at 1/4 in. H_2O)

In order to proof test the Bodega Bay pressure suppression design, Pacific Gas and Electric is conducting a test program at its Moss Landing Power Plant. As in the Humboldt Bay case, the applicant has constructed a full scale segment of the suppression system. In the test for Bodega Bay, a single 24-inch diameter vent pipe ~~xxxxxx~~ from the dry well to the suppression chamber was used. Since the full size plant is to have 112 of these vent pipes, the test equipment represents a 1/112 segment of the plant containment.

Tests were conducted with this mock-up to simulate various accident conditions. A flow comparable to 1/112th of the flow resulting from a complete circumferential break of one of the 28 in. reactor recirculation lines (with flow out both sides of the break) was taken as the "maximum credible operating accident" (MCOA). Highest containment pressures observed in these tests were 52 psig in the dry well and 30 psig in the suppression chamber. These pressures were observed when the mock-up dry well was pre-heated to 255° F and when the mock-up reactor vessel water was subcooled 35°F. Tests at higher and lower dry well temperatures and at higher and lower reactor water subcooling yielded lower dry well and suppression chamber pressures.

In another test a break area 2.5 times that of the MCOA was simulated. In this test, the peak dry well pressure observed was 63 psig. Further Moss Landing tests are being conducted to determine whether baffles are needed in the suppression chamber.

As another significant containment design feature, Pacific Gas and Electric proposes that in a ~~number of~~ some instances a single isolation valve (rather than double isolation valves at the containment) will ~~not~~ be installed at the containment wall in pipes or ducts penetrating the containment. The applicant states, however, that in these instances each such line will have an additional ~~two~~ isolation valve~~s~~, ~~one of which is~~ which is a remotely operable process valve located elsewhere ~~located elsewhere~~.

(Note: Additional remarks on isolation valving will be made after PG&E submits amendment.)

The ~~two~~ isolation valves located at the dry well wall in each main steam line are to close on a manual signal or automatically on the occurrence of any of the following:

1. Low condenser vacuum
2. Main steam line leak (in the pipe tunnel)
3. Low reactor water level.

The Bodega Bay design is such that during refueling, the spent fuel storage pool will connect directly to the shield water above the reactor, thus permitting direct underwater transfer of fuel without the need for a ~~single~~ transfer cask. This feature provides ~~in a simple and reliable way~~ for both continuous shielding and cooling of spent fuel during transfer and storage.

The refueling building, in which the drywell and suppression chamber system is located, is provided with a controlled release ventilation system which discharges to the plant stack. The building and ventilation system design is such that the refueling building can be maintained at a negative pressure while ~~at the same time~~ the ventillation discharge from the building passes through ~~cleanup equipment which removes~~ halogens and radioactive particulates cleanup equipment prior to discharge to the stack. Pacific Gas and Electric ~~PG&E~~ has indicated that, in accordance with the recommendations ~~[illegible]~~ of the ACRS, the ventillation cleanup system will be designed to permit ~~[illegible]~~ testing of the facility for filtering ~~ability to filter~~ particulates and ~~to~~ for removing ~~remove~~ iodine at specified efficiencies.

For a reactor of the type proposed, the staff believes that the general containment scheme ~~proposed~~ is adequate. We believe, however, that some important criteria for the design of the containment features have not at present been specifically proposed. Such additional criteria, including those mentioned explicitely by the ACRS report to the Commission, are necessary to assure that the containment as proposed can be reasonably expected to provide the high degree of integrity ~~proposed~~ at any time that it might be called upon to contain the ~~xxxxxxxxxxxx~~ consequences of a~~n~~ ~~maximum credible~~ accident.

These criteria involve containment testing, penetration design, and isolation valving as outlined below.

1. The design should permit initial integral leak rate testing of the dry-well and suppression chamber at their respective design pressure after the installation of all penetrations (including piping conduits, electrical conductors, and gasketing closures) and subsequent periodic testing at ~~suppression pool~~ their design pressures. In the initial testing, the leakage rate of the containment system should be determined as a function of ~~xxxx~~ pressure up to full design pressure.

2. The design of penetrations should take into account, in addition to the pressure load, the loads or deformations imposed by thermal expansion, impact of missiles, reactions of ruptured pipes, and disturbances incident to the installation, maintenance or repair of these penetrations. Penetrations should be shielded from missiles to the extent practicable. All penetrations should be designed so as to allow frequent periodic leakage rate tests of the penetrations only (including points of attachment to the containment shell), at full design pressure.

3. All pipes and conduits which communicate with interior of the primary system or the containment system, and other piping (such as instrumentation and control piping) which cannot be adequately protected against accidental rupture, should contain double isolation valves. All valves performing the function of isolation valves should be provided

with protection against materials in the system which might prevent proper closing and should be provided with reliable automatic and manual actuation features. Isolation valving should be designed so as to permit periodic leakage rate tests. Appropriate closing times for isolation valves should be determined on the basis of analyses of system ruptures which would release fission or activation products outside the dry well while the valves are not fully closed.

The Staff believes that PG&E should submit for Commission review the results of further developmental tests of the suppression pool concept and final design plans for the containment as soon as they can be made available.

(Note: Additional comments will be made on the effects of earthquakes on containment design.)

D. **Reactor Nuclear Systems**

The features of a nuclear power plant which are of most important to reactor safety are those which provide for the containment of radioactive fission products in normal operation and ~~[struck out]~~ those systems for instrumentation and control of the neutron ~~charge~~ chain reaction. Generally speaking the mechanical design of the reactor fuel elements and the design of controls and the heat removal system should be such that in normal operations and under ~~many~~ many conceivable accident situations radioactive fission products ~~would be~~ are confined to the fuel elements themselves. In this context ~~The~~ the fuel cladding comprises the primary containment system and ~~this objective is met then~~ the primary coolant system itself serves as a secondary containment system.

1. Primary System

The reactor ~~core will be situated in a reactor~~ pressure ~~vessel~~ vessel will be designed, built and tested in accordance with Section VIII of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers. The 50 ft. high by 15 ft. diameter vessel will be constructed of carbon steel approximately 6 inches thick. The interior ~~of which~~ of the vessel will be clad with approximately 1/4 of an inch of stainless steel applied by weld overlay methods. The design pressure ~~of the vessel~~ will be 1235 psig at 575°F.

~~[illegible]~~

Steam which is generated in the reactor is passed through ~~after separation~~ axial flow steam separators and steam driers located ~~from the boiling water~~ inside the reactor vessel ~~will to~~ ~~and then~~ through ~~=~~ two 20-inch steam lines to the turbine which is located in a separate structure. Water from the steam separators drains to the downcomer annulus ~~after separation from steam in the reactor vessel~~ where it passes ~~from the reactor vessel through~~ to four 28-inch recirculation pipes. Four recirculation pumps, one in each loop, provide the driving force for circulating water through the reactor core. Feed water ~~returning~~ from the condenser is injected into the reactor vessel by a pump driven by the main turbine ~~shaft~~.

~~Pressure vessel and~~ The piping located within the dry well will be designed, tested and constructed in accordance with applicable requirements of the ASME Boiler and Pressure Vessel Code. Piping outside the dry well will conform to the requirements of the American Standards Association Code for Pressure Piping. Twelve safety valves, arranged to discharge into the suppression chamber, are provided to protect the reactor and primary system from over-pressure.

2. Core Design

The reactor core will be composed of 592 fuel assemblies each of which provides a vertical channel through which the mixture of steam and water passes.

The core will have approximately the form of a right circular cylinder 140 inches in diameter and 125 inches high. One hundred and forty-five control rods will enter the core from below the fuel assemblies ~~which~~ through control rod guide tubes. ~~[illegible]~~

~~[illegible]~~

~~[illegible]~~

~~[illegible]~~

The fuel assemblies rest on fuel support plates ~~form guides for the movement of control rods in the~~ which are supported by the control rod guide tubes. ~~core, are held in proper position by upper and lower~~ Fuel alignment is provided by upper and lower grid plates which are attached to the cylindrical core shroud. ~~The weight of the fuel assemblies is borne by the control rod guides which extend to the lower head of the reactor pressure vessel.~~ (Note: Conclusions on structural design will be stated as soon as seismological reports have been received and evaluated.)

Each fuel assembly will be composed of 49 fuel rods in a square array. Fuel pellets of UO_2 enriched to 2.5% U-235 will be contained within stainless steel tubing. ~~PGAE~~ Pacific Gas and Electric has tentatively proposed that this tubing would have a nominal thickness of 0.011 inches and would be able to withstand an exposure of 15,000 MWD/TON. On the basis of present information and operating experience, one cannot be sure that fuel with this type of cladding can be irradiated for the exposures stated without experiencing extensive rupturing of ~~xxxxx~~ cladding. The ~~fuel~~ design will not be ~~finally designed~~ completed until further data from a General Electric research and development program are available. This program is ~~necessarily~~ designed to provide an engineering basis for the fuel element design selection. In any event, extensive experience with other power reactors provides reasonable assurance that a ~~safety~~ suitable design can be ~~made.~~ developed.

Thermal and hydraulic factors, which ultimately determine the permissible power level of the reactor, ~~has~~ have only been briefly described by PG&E. The data presented ~~in the case~~ indicate that PG&E intends to extend the limits generally in use in reactors of this type. Analyses made at this time, ~~xxxxxx~~ together with available operating experience, do not form a sufficient basis for determining appropriate thermal limits at this time. PG&E has had established ~~b~~ design criteria for establising ~~a~~ detailed thermal and hydraulic design factors they will use in the Bodega reactor, namely, that the fuel will operate without loss of cladding integrity over the design exposure period at the maximum heat fluxes possible within burn-out limitations. Operating experience at other boiling water reactors has indicated that this criteria can be met.

May 27, 1963

The power distribution which is expected in the Bodega Reactor core has been estimated for the purpose of determining the thermal margins which would be ~~obtain~~ attained in the hottest channel ~~fuel element~~. These estimates will be refined by detailed calculations of power distribution in the course of the design ~~of the reactor and in operation the power distribution~~ and will be monitored continuously by a system of incore flux monitors during subsequent operation. Such ~~systems~~ methods have been successfully used in other reactors and should provide ~~should provide a reliable~~ the means for operating within the ~~of establishing the~~ established thermal margins ~~that are experienced in operation~~ for this reactor.

Preliminary calculations indicate that at rated operating conditions the steam volume fractions would be as follows:

Average Core Voids-37%

Average Exit Voids-58%

PG&E believes that analog computer studies which being made will show that the plant can be designed to exhibit satisfactory dynamic performance with such high void content. ~~The~~ The staff knows of no substantial operating experience that would confirm the acceptability of operating at void fractions this high; ~~It is clear,~~ however, ~~that~~ with appropriate limitations high void conditions can be safely approached in reactor tests

which could maximum
~~devised so as to~~ determine the ~~proper range~~ of void content for normal operation.

3. <u>Reactor Controls</u>

Nuclear safety requires that there be reliable means for controlling the reactivity of a nuclear reactor. Reactivity can be considered a measure of the rapidity with which the neutron chain ~~xxxxxxx~~ reaction changes. When reactivity is positive the chain reaction grows, that is the rate at which nuclear energy is released by fission is increased. Conversely when reactivity is negative the chain diminishes and power falls. omit

The operating condition of the reactor, its temperature, pressure, power level, void content~~s~~ and ~~age~~ fuel exposure all ~~have an~~ ~~effect on~~ reactivity. The general way in which each of the above parameters effects reactivity are quite ~~nature of these effects xxxxxxxxx is quite~~ well known, and the theoretical and experimental methods for investigating the reactivity effects are sufficiently developed to permit design of the ~~xx~~ reactor control system to proceed with confidence.

no ¶ It is expected that within the range of operating variables contemplated for this reactor the reactor should ~~xx~~ be stable; That is an increase in reactor power ~~any changes in operating~~ causes changes in the operating parameters which ~~xxxxx condition which accompany increases in power~~ have a strong tendancy to decrease reactivity and tend to limit ~~The~~ the power increase. The main purposes of the control system, therefore, are to provide a means of precisely adjusting reactivity to control the reactor operating parameters ~~and reactor power~~ and to provide a fast and reliable means of terminating reactor operation ~~a chain reaction~~ should any condition occur in the system that ~~accidents cause a rapid power excursion xxxxxxxxx~~ could if allowed to proceed unchecked result in the ~~which could as a consequence~~ release of fission products from the fuel elements or might damage the primary system.

~~The principal feature of~~ The control system proposed for this reactor is an array of 145 movable ~~the array of 145~~ control blades which ~~has~~ sufficient reactivity worth to keep the reactor safely shut down even though one of these blades ~~rod~~ might be stuck ~~out of the~~ in the reactor core.

~~The reactor will have a cold, clean, uncontrolled keff of 1.27.~~ The keff of the reactor with all control blades ~~rods~~ in the reactor core is calculated to be 0.97. The combined worth of the 145 control blades ~~rods~~ is calculated to be 0.18. The control material will be boron carbide contained ~~in~~ within 0.175 in. O.D. stainless steel tubes. Additional control is provided with 316 fixed control curtains which will be semi-permanently located between selected fuel elements. The worth of these curtains is calculated to be ~~xxx~~ 0.12. These ~~The~~ control curtains will be constructed of 0.1% boron stainless steel.

May 27, 1963

The reactor design also incorporates a liquid poison system that can be used to inject ~~xxxx~~ sodium pentaborate into the core in the event complete shutdown cannot be achieved by use of the control rods.

~~The~~ Hydraulic control rod drives are proposed to ~~be used in the Bodega Bay plant are to be~~ position the control blades within the reactor core. These drives are to be designed using the same basic concepts as have been employed in drives in use in boiling water reactor plants at Dresden, Big Rock Point, Humboldt Bay and the SENN Plant in Italy. Water ~~uses~~ is proposed as the hydraulic fluid and can be applied to either side of a piston which is mechanically coupled to the control rod, thus providing for either upward or downward rod motion. Only one rod can be moved outward (increasing reactivity) at a time and it may be moved either continuously or in 6 inch steps. Rod speed is controlled by orifices which regulate the flow of water away from the low pressure side of the piston. All rods can be inserted simultaneously, shutting the reactor down. Rods are scrammed upward by applying pressurized water from either the reactor or from accumulators to the bottom side of the drive pistons and simultaneously relieving the ~~xx~~ volume above the top ~~xxx~~ side of the pistons to the scram dump tank. The drive is locked in fixed positions by collet fingers which engage grooves spread at 6-inch intervals along the moveable index tube. The collet fingers support the weight of the rod and the downward forces due to reactor pressure.

Since drives similar to these have been used at other plants an important part of our evaluation of these drives is based on previous experience with these drives. ~~This includes Dresden experience as well as initial Big Rock operations.~~

no ¶ At Big Rock Point, there have been two isolated occurrences of rod "drift-out". In one of these, the cause was attributed to an inadvertent release of demineralizer resins resulting in the collet fingers being jammed in the open position so that the rod was free to drift as influenced by the forces due to gravity and hydraulic pressure. In the second case it was reported that a hard particle became trapped between the collet piston and a sleeve which is located between the collet and the index tube. This again is believed to have caused the collet fingers to be jammed in the open position, thus permitting rod drift. The hard particle was never found. It should be noted, however, that in neither of these cases nor in any other case has there been any apparent significant impairment of scram capability. Detailed design of drives for the Bodega reactor has not been made. General Electric is considering modifications of earlier designs that will minimize the possibility of foreign material accumulating in the rod drives. The applicant has also indicated that functional and endurance tests will be made on the prototype Bxxxxxxx Bodega mechanisms, but the detailed procedures for these tests and the acceptability criteria have not been determined.

Discussions of the possible consequences of a rod dropout accident involving a single rod is discussed elsewhere in this report under Accident Evaluations. It is our opinion that the sudden dropout of more than one rod at any given time is incredible with ~~thmp~~ the proposed system.

check on this

Control systems which are designed to react rapidly to demands for shutting a reactor down generally have some potential for accidentally increasing reactivity as well. This aspect of the PG&E control system design is discussed later in this report (Section V), where consequences of a rod drop-out are considered. PG&E has indicated that devices for limiting individual rod worth and for impeding the fall of a rod are under development. Such devices could enhance the safety of operation and simplify the procedures that are presently used with similar drives to provide assurance that rods cannot drop out.

check on this also

The staff believes that PG&E should submit timely reports to the Commission on development, design, and testing of the control system.

(Note: Additional comments will be made on earthquake effects.)

4. Control and Safety Instrumentation

The instrumentation necessary for safety in a nuclear power plant generally involves a large number of sensors throughout the various process systems. These sensors measure a variety of variables, such as neutron flux and gamma radiation levels, and temperatures and pressures of various fluids. Information collected by the measuring instruments is used to guide the operating staff in controlling the plant and to actuate automatic control devices.

The instruments, circuits, and control devices which are of most importance to public health and safety are: (a) those necessary for and contributing to stable reactor operation, (b) those used in control of radioactive fluids and effluents, and (c) those used for control of emergency equipment.

A general description of instrumentation is presented in the application. At this time, there is not sufficient information available from which one can determine whether instrumentation provisions have been made for all essential functions, nor can one determine the degree of reliability that should be attributed to the reactor protection system, which is described by PG&E as "fail-safe". These, however, are design problems which appear to require only the application of well-known engineering methods. The staff intends to evaluate the reactor control and safety instrumentation in detail. Particular attention will be given to the need for automatic functions and the reliability of safety instrumentation.

(Additional comments will be added on earthquakes.)

E. Emergency and Safety Systems

Emergency systems provide means either for safely continuing operation in the event of some equipment failure or operator error or for limiting the extent of damage and resultant hazard. In many instances, design features of the facility which have been provided for the primary purpose of making plant operation more convenient, reliable for economic or, in effect, emergency systems. Other features are designed primarily as emergency systems.

The principal emergencies facilities proposed are: (1) Alternate power supplies for critical electrical loads; (2) Reactor control safety devices and circuitry; (3) Liquid poison injection systems; (4) Emergency cooling system; (5) Bleed and feed system; (6) Core spray system; and (7) Containment system.

Some of these systems have already been discussed in this analysis. Principal features of other emergency and safety systems are discussed in this section.

In all such systems one must require a high degree of reliability of the system to perform properly in adverse circumstances. This requires not only careful design of the outstanding features but also attention to such related equipment as signal and control circuits, power supplies, and ~~in~~ instrumentation. Maintenance and frequent testing of emergency systems ~~must~~ provides the final assurances of readiness of emergency systems to respond to the demands placed upon them; these factors must, therefore, be taken into account in final design.

1. <u>Power Supply</u>

Protection of power supplies is provided on several levels, as described in PG&E's ~~[illegible]~~ application. The plant is tied into the PG&E distribution system by two 220 Kv circuits to Ignacio Substation. All plant auxiliary power requirements can be met by either a transformer tied to the station generator or by a transformer tied to the 220 Kv lines. An additional external transmission line and transformer of limited capacity and an engine-driven generator provide emergency power to equipment necessary for safe plant shut-down. Station batteries will supply the electrical energy for the more critical loads.

2. <u>Emergency Cooling Systems</u>

A number of ~~[illegible]~~ different systems ~~means~~ will be provided for removing ~~the heat~~ after-heat generated in the reactor core as a consequence of ~~by the [illegible]~~ radioactive decay of fission products. Such provisions are necessary to remove the decay heat after reactor shutdown to prevent melting or rupture of fuel elements (which would lead ~~and the~~ to the release and dispersal of fission products. These provisions will include:

(1) The normal condensate-feed-water system;

(2) An emergency condenser which can be put into operation in event the reactor must be isolated from the main condenser;

(3) A low pressure shutdown cooling system; ~~and~~

(4) A bleed-and-feed system which releases steam at a controlled rate to the suppression pool.

(5) a high pressure core spray system and (6) a low pressure core spray system

A number of sources of water (and pumping capacity) will be available to restore water lost through accidental ruptures or through bleed-and-feed operations. Both high head and low head pumps will be provided with back-up pumping arrangements. In the event of a major rupture of the primary system, emergency action should be capable of reducing to a great extent the amount of fuel damage and fission product release from the reactor.

(Additional comments will be made on earthquakes.)

V. Safety Analyses

The design features of theplant have been described in the previous sections and in many cases the safeguards provided by a particular design feature or the operational limits imposed by safety considerations for a particular feature were discussed. In general the criteria for plant design should include: (1) means to control radiation hazards (including radioactive discharge) during normal operation, (2) design features to minimize the probability of having an accident, and (3) design features for mitigating the consequences of an accident should one occur.

The means for controlling radiation hazards during normal operation will be provided by suitable shielding and radiation monitoring in the case of direct radiation emitted from the reactor and by proper monitoring of radioactive wastes which are discharged from the plant site. For wastes discharged from the plant, the release rates shall be such that they do not result in personnel exposures in excess of 10 CFR 20 limits.

The adequacy of the design features that are incorporated to mitigate the condequence of an accident in the unlikely event that one should occur are evaluated in the following section on the maximum credible accident. The consequence of this accident to the health and safety of the public is presented taking into consideration the safety features afforded by this containment system and the environmental character of the site.

To evaluate the design features that are incorporated into the plant design to minimize the probability of having an accident a number of representative abnormal conditions, equipment malfunctions and operator errors were postulated and evaluated by the applicant. Those which were presented in the Preliminary Hazards Summary Report included:

a. Changing pressure regulator handwheel setting

b. Continuous control rod withdrawal or insertion

c. Loss of electrical load

d. Control rod drive malfunction

e. Recirculation pump failures

f. Main steam valve closures

g. Failure of reactor safety valve to reseat

h. Failure of reactor safety system

i. Fuel cladding failure

j. Loss of feedwater

k. Loss of condenser vacuum

l. Loss of auxiliary power

m. Instrument air failure

n. Pressure regulator failure

o. Emergency condenser tube failure

p. Reactor system ruptures inside the dry well

q. Failure to replenish cooling water in emergency condenser

r. Startup accident

s. Fuel loading and handling accidents

t. Cold water accident.

In addition to those conditions listed above, three equipment failures termed "Major Accidents" were evaluated by the applicant. These accidents included:

~~Six~~

a. Control rod drop accident

b. Main steam line rupture outside the dry well

c. Reactor system rupture in the dry well.

In some of the malfunctions and failures presented, the evaluation is not yet completed, however, the applicant has stated that when the analysis is complete, the results will be used as criteria in the detailed plant design (for example, to size the pressure relief valves and to set the isolation valve closure specifications). In our opinion, the evaluation results which are complete and the stated design objectives for plant systems and components where evaluation is incomplete are satisfactory ~~with one exception. In the "Control Rod Drop Accident" calculations by the applicant indicate that the most reactive control blade could have a reactivity worth as high as .036. Additional calculations show that if this blade were to drop free of the core~~

with one exception. In the "Control Rod Drop Accident" calculations by the applicant indicate that the most reactive control blade could have a reactivity worth as high as .036. Additional calculations show that if this blade were to drop free of the core a minimum period of 3 milliseconds could result, and the average fuel temperature would reach 5500°F in the uncontrolled fuel zone. The consequences to the reactor vessel in the event of this accident are not entirely clear. The applicant has indicated they are developing analytical models for more accurate prediction of the consequences of such a nuclear excursion and that the forthcoming SPERT destructive test will be used to check the model that is developed.

In addition to the analytical work, a rod worth minimizer computer and a rod dropout velocity limiter are being developed for possible use in the Bodega Plant. The rod worth computer would continually monitor control rod patterns to reinforce procedural controls provided to insure that patterns causing individual rods to assume undesirably high reactivity worth are not used. Conceptual designs for flow restricting devices that would limit potential control rod dropout velocities to safe values are also being developed. In the absence of experimental verification of the applicants position that a rod dropout accident of this type will not endanger the reactor vessel, we believe that other design features, such as the rod worth minimizer computer or the rod dropout velocity limiter, should be incorporated into the plant design.