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DEEP—POOL REACTOR
FOR
WATER DESALTING

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DEEP—POOL REACTOR
FOR
WATER DESALTING

U. S. ATOMIC ENERGY COMMISSION
CHICAGO OPERATIONS OFFICE
CONTRACT NO. AT(11-1)- 1444
JUNE 30, 1965

Prepared by:
BECHTEL CORPORATION
SAN FRANCISCO, CALIFORNIA

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SUMMARY

This report presents the results of an engineering design study performed by Bechtel Corporation for the Chicago Operations Office of the Atomic Energy Commission to investigate the potential and feasibility of a deep-pool reactor concept for ultimate use as an economic heat source for a single-purpose water desalting plant. The work performed under this contract consisted of a refinement of the design and costs developed for this concept by the Chicago Operations Office.

Major effort was concentrated on the conceptual design of the reactor portion of the plant; however, the unique features required for the water plant were identified and a preliminary optimization of the overall facility was made. Cost estimates were prepared for the reactor plant and for the total facility at a typical southern California site and the cost of heat and water was estimated. Conceptual designs and estimates were prepared for plants of three sizes - 10, 50, and 200 mgd. The 50-mgd plant was the base case. The design and cost estimates for the 10- and 200-mgd plants were made by adjustments from the base case.

The reactor consists of a conventional water reactor core located in the bottom of a 150-foot deep tank; the tank is located within a concrete structure with the reactor core approximately 90 feet below grade. The free water surface at the top of the reactor tank is essentially at atmospheric pressure; hence, all pressurization required to prevent boiling in the core is provided by the head of water above the core. The bulk of the water in the tank is at 200°F (the core inlet temperature) and the coolant flow is downward. Pumping head is provided by vertical centrifugal pumps having sufficient rotating inertia to provide safe flow coast-down in the case of loss of power.

The brine heaters for the multistage flash evaporator plants are oriented vertically and located in a pit adjacent to the reactor tank. Since the reactor coolant outlet temperature is 270°F, a more efficient thermodynamic system is obtained by dividing the water plant into three separate units, each operating at a different maximum brine temperature. Based upon preliminary optimization of the performance ratio of the water plant, the reactor power levels required to produce the 10-, 50-, and 200-mgd water output are 70, 400, and 1600 Mwt, respectively. Brine side pressures are maintained slightly higher than the reactor coolant pressure and double tube sheets are used to minimize the possibility of contamination of the brine with reactor coolant.

The physical arrangement of the reactor eliminates the possibility of uncovering the core in the event of a loss of coolant accident. Further, the large volume of the primary system has a substantial thermal

capacity which will provide cooling for the core for many hours in the event of loss of flow or system rupture. These and other inherent safety characteristics make it quite reasonable to assume that the very low leakage containment vessels normally provided for water-cooled power reactors would not be required for the system.

The cost estimates developed in this study were based on the assumption that the plants were "second of a type" units. In addition, it was assumed that these plants will be built on a turn-key basis with a single A-E responsible for engineering design, procurement and construction. Fuel costs were based upon present day numbers and substantial future improvement should be expected in these.

Total field costs for the three reactor plants excluding engineering, contingency and fee were as follows:

70-Mwt Reactor Plant	\$ 3,990,000
400-Mwt Reactor Plant	5,970,000
1600-Mwt Reactor Plant	13,600,000

The corresponding total plant costs, including an allowance for land and owner's costs, were as follows:

10-mgd Plant	\$ 21,800,000
50-mgd Plant	53,400,000
200-mgd Plant	178,200,000

It can be seen from the above results that the cost of the water plant is by far the largest portion of the total plant cost.

The total cost of heat for the three reactors is:

70 Mwt	80¢/10 ⁶ Btu
400 Mwt	32¢/10 ⁶ Btu
1600 Mwt	24¢/10 ⁶ Btu

The corresponding cost of water for the three plants is:

10-mgd	91¢/1000 gal.
50-mgd	50¢/1000 gal.
200-mgd	42¢/1000 gal.

The estimates also show that the nuclear fuel costs account for from approximately 30 percent of the cost of heat for the 70-Mwt reactor to over 65 percent of the cost of heat for the 1600-Mwt reactor; and that total energy costs account for from approximately 50 percent of the cost of water for the 10-mgd plant to somewhat less than 40 percent of the cost of water for the 200-mgd plant. Perhaps it is more significant to note that the reactor plant capital cost contributions to the total cost of water vary from a low of 5.5 percent for the 200-mgd plant to a high of 17.4 percent for the 10-mgd plant. Thus, the capital cost of the heat source is a rather small portion of the total plant costs for all cases studied. The reduction in cost of water in going from 50 to 200 mgd is less than 20 percent. Thus, it appears that single-purpose water conversion plants of this type in the range of 50 to 100 mgd have reached the "knee" of the unit cost curve and very small improvements in product costs would be expected from larger plants.

Future improvements in water costs would appear to be possible as a result of reductions in fuel and capital costs. Fuel costs are expected to fall significantly in the next few years. Capital cost reductions are possible as the result of future improvements in water plant designs and more favorable site conditions. The combined effect of these reductions could result in a decrease in water costs of as much as 10 to 20 percent.

Section 1

INTRODUCTION

In view of the widespread interest in developing practical processes for desalting saline water, particularly those processes utilizing the economies of large-scale nuclear heat sources, this study was undertaken to determine the technical features and the economics of a particular reactor concept offering the potential of low capital and energy costs for use in a single-purpose water desalting plant. In order to evaluate the economics of the system, the deep-pool reactor heat source was assumed to supply heat to a multistage flash evaporation water desalting plant. The emphasis of the study was on the reactor heat source. The design and cost estimate of the water plant itself was based on previous studies and on a computer code for optimization of multistage flash evaporation plants. The components of the water plant were identified only to the extent necessary to be compatible with the unique features of the deep-pool reactor.

The reference case is a 50-million gallon per day (mgd) plant. The plant description and detailed cost estimate in this report refer specifically to this case. Adjustments were made to plants of 10-mgd and 200-mgd capacity to provide an indication of the effect of size on the cost of the water produced.

Special consideration was given to the unique aspects of the reactor plant design, including material compatibility and the design requirements associated with the depth of the reactor tank. The safety advantages inherent in the essentially unpressurized, deep-pool reactor concept were utilized in the plant design and were evaluated in a preliminary accident analysis. Siting studies were made to define the site conditions to be assumed and the plant design features required for the hypothetical southern California coastal site assumed as a basis of the study. Auxiliary systems comparable to those in existing water reactor power plants were defined.

The conceptual design of the reactor plant and the associated water plant is based on the following ground rules:

1. The design is based on existing technology to the maximum practical extent. No credit is taken for future technological developments or future decreases in the cost of fuel, materials or components, or for less costly sites. Technical uncertainties are identified and, where considered necessary, the need for proof testing is indicated. It should be emphasized, however, that the conceptual design presented in this report is considered to be technically feasible and capable of being

constructed using present-day technology. It is expected that future developments would tend to further improve the concept and to reduce its cost.

2. The plant is located at a hypothetical site along the coast of southern California. The site conditions assumed are representative of those at suitable sites available in this area. No attempt is made to assess the demand for water in this area or to compare the relative merits of this concept with other nuclear-fired single-purpose water plants, with fossil-fuel-fired single-purpose plants, or with dual-purpose power and water plants.

The cost estimate of the plant is based on the following ground rules:

1. The plant is considered to be the second of its type; that is, a previous plant has been built to essentially the same design and all necessary proof testing has been performed.
2. A single architect-engineer, assumed to be prime contractor, is responsible for the design and construction of both the nuclear plant and the water plant, under a turn-key arrangement. In this arrangement, the prime contractor is responsible for engineering, procurement, construction, testing, startup, and all other services, except those provided by the client, necessary to obtain a complete and operable plant.
3. Energy and product water costs are based on currently-quoted nuclear fuel costs, operation of the plant at 90% capacity factor, a fixed charge rate of 7%, purchase of the necessary electrical power to operate the plant at 6 mills per kilowatt hour, and other operating and maintenance costs typical of plants of this type.

This study was an outgrowth of a preliminary study made by the Chicago Operations Office (COO) of the U. S. Atomic Energy Commission in which the deep pool-type reactor concept was devised and shown to have potential for desalting applications. Results of the initial study are reported in COO-278, "Conceptual Design of a Reactor for Use in Desalination", October 1964.⁽¹⁾ In the current study, COO was responsible for the reactor core and directly related items, and provided conceptual design and cost information for the following: fuel, core configuration and structure, reactor shielding, control rods, control rod drives, refueling equipment, and nuclear instrumentation. COO also performed a preliminary thermal-hydraulic analysis of the core. The conceptual design and cost estimate for the remainder of the total plant was Bechtel's responsibility. For the sake of clarity and continuity, the information provided by COO has been integrated into this report. Most of this information is included in Paragraph 5.2.

Section 2

GENERAL PLANT DESCRIPTION

The deep-pool reactor plant described in this report was conceived specifically as a source of process heat for a single-purpose water desalting plant using the multistage flash evaporation process. The reference plant has a capacity of 50 mgd with a reactor heat output of 400 Mwt. Section 9 briefly describes plants with capacities of 10 mgd and 200 mgd using the same design concept and with reactor outputs of 70 and 1600 Mwt respectively.

The plant is assumed to be located along the coast of southern California. Site conditions considered to be "typical" of the available and suitable sites in this area impose some penalties on the design and cost of the plant. Example of these conditions are: the relatively high population density throughout the area; the high seismic activity; the occurrence of earthquake faults which precludes the location of plants in areas which might be otherwise attractive; the high cost of land nearly everywhere along the coast; and the need for lengthy off-shore brine intake and discharge lines. Considerable excavation is required to prepare the site since the water plant requires a large, relatively level area located near sea level and since much of the available coastal area consists of bluffs rising steeply from the ocean. The deep pits required for the reactor plant must be excavated in fairly well-consolidated material to depths considerably below sea level.

The principal feature of the reactor plant is the 150-foot deep tank of high purity water in which the reactor core is located. The reactor tank is contained in a steel-lined concrete pit extending from 100 feet below grade to approximately 50 feet above grade. The water in the reactor tank provides sufficient static head to prevent boiling in the core at a reactor coolant outlet temperature of 270°F. Reactor coolant flows downward through the core, passes through three stages of brine heaters in series, and returns to the reactor tank above the core at a temperature of 200°F. The reference plant design has a single circulation loop with two parallel half-capacity main coolant circulating pumps located at the reactor outlet. The brine heaters are located entirely below grade in a 100-foot deep pit adjacent to the reactor pit.

The plant design is determined largely by the reactor outlet and inlet temperatures. The outlet temperature is limited by the maximum temperature to which brine can be heated in the brine heaters. Since water plant efficiency increases with increasing brine temperature, practical operating limits are normally set just below the temperature at which scaling of the brine heater tubes becomes a problem. Based on current

technology, a maximum bulk brine temperature of approximately 250°F was assumed; the reactor outlet temperature of 270°F was based on this brine temperature. The 140-foot head of water in the reactor tank provides sufficient thermal margin in the core at this temperature.

The reactor inlet temperature is kept below atmospheric boiling temperature. This minimizes stored energy and vapor loss in the reactor system and yet results in an efficient water plant. Convective heat transfer will maintain the column of water in the reactor tank above the core at essentially reactor inlet temperature during operation. Since the incoming brine must have a temperature of approximately 190°F or less to cool the reactor coolant to 200°F, the total brine temperature rise in the brine heaters is about 60°F. However, since the optimum brine temperature rise in a brine heater is approximately 10°F, three stages of brine heaters are arranged in series to provide the total reactor coolant ΔT in three increments. Each brine heater stage is connected to a separate, optimized water plant. The first water plant or high-temperature stage is the most efficient and produces nearly half of the total water output. Two parallel brine heaters are included in the first stage and one each in the second and third stages.

With the exception of the tubes in the brine heaters, the main coolant system and all auxiliary reactor systems are made of carbon steel. Brine heater tubes, comprising a large proportion of the total surface area in the main coolant system, are made of 70-30 cupro-nickel. Corrosion of reactor system materials is minimized by providing a cap on the reactor tank to exclude air and by proper coolant chemistry control utilizing the main coolant purification system. In addition, a coolant deaeration system is provided for any required de-gassing of the main coolant to remove air, uncombined hydrogen and oxygen from radiolytic decomposition of water, and any gaseous fission products which may be dissolved in the coolant.

To eliminate the possibility that the product water could become contaminated with radioactivity, the brine is maintained at a higher pressure than the reactor coolant in the brine heaters so that any leaks which might develop in the brine heaters would result in leakage of brine into the coolant rather than the reverse. Since brine contamination of the coolant would rapidly deplete the ion exchange resin in the coolant purification system, the probability of brine heater leakage is minimized by providing double tubesheets on the brine heaters. Similarly, all other reactor system heat exchangers in which heat is transferred directly from reactor grade water to sea water are provided with double tubesheets.

The reactor fuel is Zircaloy-clad UO_2 similar to that used in present-day power reactor cores. Core parameters typical of current water-cooled power reactors were assumed and no attempt was made to optimize

the core design for the particular application or to take advantage of possible improvements in fuel cycle costs due to the low temperature operation of this core. The core is supported from the reactor tank on a circumferential flange located near the bottom of the tank. Lead and steel shields surrounding the core inside the reactor tank and the large annulus of reactor coolant around the core provide sufficient shielding to reduce radiation heating of the concrete in the reactor pit to a negligible level. The control rods are actuated by control rod drive mechanisms mounted on the reactor tank lid at the top of the tank. The drive mechanisms are connected to the control rods by control rod extension pipes approximately 130 feet long. Refueling is performed with the assistance of closed circuit TV from the top of the reactor tank; a fuel removal tool and a fuel hoist are used to bring the fuel elements up near the top of the reactor tank. The spent fuel pit adjacent to the reactor pit is connected to the reactor tank with a refueling trough during refueling operations so that all fuel handling is performed under water.

The reactor pit and brine heater pit are constructed by sinking a single caisson from grade down to the 100-foot depth required. Concrete is then poured to separate the two pits and to provide shielding between the reactor tank and the equipment located in the brine heater pit. The total free volume of both pits is such that in the event of a main coolant system rupture, the coolant would reach an equilibrium level well above the top of the core so that fuel melting would not occur. Even if the pit walls should rupture, the core is well below sea level so that water would flow into the pits rather than out of them. The brine heater pit is covered with removable interlocking concrete beams to provide radiation shielding of the brine heaters during reactor operation. A steel cap covers the reactor pit and provides some shielding.

The reactor pit cap and brine heater pit shielding beams also are low-leakage barriers to permit controlled ventilation of the pits. Ventilating air flows from the reactor building, into the brine heater pit and the reactor pit, through a filter system to the stack. Flow is such that any leakage is inward; any radioactive contamination in the pits will not spread to other areas of the plant. The reactor building, which includes the high bay area over the reactor pit and spent fuel pit, the cover over the brine heater pit, the radwaste vault, and the auxiliary equipment space, also has controlled ventilation exhausting up the stack, along with provision for filtering the exhausted air if necessary. The reactor building and its controlled ventilation system provide a system of containment for fuel handling operations (in addition to the barrier provided by underwater refueling), as well as provide a secondary containment barrier (in addition to the barrier provided by the pits) for the reactor system.

Several auxiliary reactor systems are provided to permit safe and flexible operation of the reactor plant. The reactor coolant purification sys-

tem maintains the reactor coolant at a high purity and controls its pH to minimize corrosion of system surfaces. The coolant deaeration system provides for spraying reactor coolant at a controlled rate into the top of the reactor tank where a portion of it will flash to steam and release dissolved gases. The space above the water in the reactor tank can be vented through a condenser to the radioactive waste disposal system.

The radwaste system collects, and processes for disposal, radioactive liquids and gases generated in the plant. The amount of waste generated during normal operation is expected to be minimal but sufficient capacity is provided to handle a wide range of conditions. For example, two 100,000-gallon liquid hold-up tanks are included in the design to permit retention of brine, which may be discharged from a brine heater in the event of a brine heater tube leak. At the same time, the tanks permit discharging reactor coolant from the defective brine heater so it can be repaired quickly. The brine is held up just long enough to check for radioactivity prior to disposal; reactor coolant may be stored temporarily for possible re-use or for treatment prior to disposal. An emergency poison injection system, consisting of both a gravity feed tank located above the reactor tank and a storage tank with a pump feed, is provided for emergency shutdown of the reactor. A decay heat removal system is provided to maintain the coolant at a low temperature for refueling. Initial cooldown will ordinarily be accomplished by circulating reactor coolant through the brine heaters. In an abnormal situation, the decay heat removal system can be used to keep the reactor coolant from boiling immediately after shutdown if heat cannot be removed in the brine heaters. The spent fuel pit water is cooled and purified with a spent fuel pit cooling and clean-up system. A sea water cooling system takes water from the water plant intake and circulates it through numerous coolers in the reactor plant, including the purification system non-regenerative heat exchanger, the spent fuel pit heat exchanger, the decay heat exchanger, the off-gas condenser, the brine heater pit air cooler, and the heat exchanger for the fresh water component cooling loop used for lube oil cooling and cooling of various other reactor plant components.

The deep-pool reactor concept and the conceptual reactor plant design developed have several safety features which are significant to reactor hazards and siting analysis. Since the reactor system is essentially unpressurized, the reactor coolant contains relatively little stored energy so that, in the event of a main coolant system rupture, only a small part of the coolant will flash to steam. This suggests that a conventional pressure-containing containment structure is not required. Since the total volume of reactor coolant is large and a significant fraction of it is below 212°F, there is an appreciable interval of time before boiling will occur even with complete loss of coolant flow. In an emergency situation where no other means of heat removal can be provided, the coolant can be allowed to boil. This should prevent core damage and

the steam generated would be directed to the stack. Since the entire main coolant system is contained in pits located well below grade and below ground water level, a complete loss of coolant accident is not considered possible and gross fuel melting cannot occur. Thus, no significant amount of radioactivity would be released from the fuel even under a severe accident condition. The controlled ventilation systems in the reactor and brine heater pits and in the reactor building provide means of confining any radioactivity that might be released so that it is filtered and released to the stack.

The electrical system provides alternate sources of power to critical components in the reactor plant. The main power supply coming into the plant provides power for operation of the water plant and for operation of the entire nuclear plant. A completely independent incoming line provides back-up power for operation of portions of the reactor plant in the event of loss of the main power supply. An emergency diesel generator provides power to maintain the reactor plant in a safe condition and to permit orderly shutdown and decay heat removal. An emergency battery supplies power for operation of nuclear instrumentation and reactor control if all other sources fail.

Section 3

SITE DESCRIPTION

3.1 GENERAL

The plant is assumed to be located at a hypothetical site along the coast of southern California. There is no one predominant topography and geology typical of the California shoreline from the mouth of the Santa Maria River southward to the Mexican border. Instead, the coastline varies from broad sandy beaches to sheer cliffs. Alluvial stream beds (valleys) and marshlands occur in numerous locations along the southern California coast. However, when availability of suitable real estate is considered, the cliff-type site probably is most descriptive of a typical southern California shoreline, and this type of site is used as the basis for the conceptual design analysis of this report.

3.2 LOCATION

No exact location has been assumed. In general, however, available sites are less than five miles from a modest-sized community and within 50 miles of a large city, viz, Los Angeles, San Diego or Santa Barbara.

Any candidate location is near a major highway and railway. At most sites, U. S. Highway 101 (four lanes typically) passes within a quarter mile of the site. A main line railroad also may pass within a mile of most available sites.

3.3 GENERAL TERRAIN

The southern California coastal area is characterized by a gently sloping coastal plain extending seaward from uplands some miles inland. At the typical site, this plain is terminated abruptly at the shoreline by high seacliffs which have been straightened over considerable distances by marine erosion. These cliffs range in height from 60 to 90 feet above sea level and are separated from the ocean by, at most, a narrow band of beach sand. In general, tidal action is limited to a gently sloping ledge of sandstone, siltstone or other soft rock. The cliffs may be deeply scored by ephemeral streams to form barrancas, or steep-sided ravines.

3.4 GEOLOGY

The geology of the typical site is represented by the geologic cross-section shown in Figure 3-1. The surface is Pleistocene Terrace consisting largely of horizontal, crudely stratified deposits, mostly fluvial but partly marine in origin.⁽²⁾ This formation constitutes the upper 30 to

40 feet of the coastal plain at the shoreline and is underlain typically by an upper Miocene marine formation or the middle Pliocene San Mateo formation, either of which should extend downward for several hundred feet. A few feet of Pleistocene littoral deposits of well-rounded gravel and sand may be interbedded between the upper Pleistocene strata and the underlying more consolidated formation.

The surface formation of Pleistocene age is composed chiefly of sedimentary deposits which exhibit wide lateral stratigraphic continuity. This formation is well weathered and can be readily excavated with earthmoving equipment such as scrapers, dozers and shovels. This formation is relatively stable and will stand unsupported at a very steep angle.

Lower strata are unweathered and of more consolidated material than the Pleistocene deposits. Typically, the underlying upper Miocene marine formation consists of a well-consolidated sandstone, siltstone, shale or conglomerate. These materials exhibit high density and strength, and they should serve as excellent foundation material with a bearing capacity of about 10 tons per square foot. Generally, this formation has an upper boundary at about elevation +20, which is an appropriate foundation level for most structures in the reactor facility and the adjacent water plant. Excavation of the upper Miocene strata will require different techniques. Massive excavation probably will require the use of explosives to break up gross quantities of the formation, but shaft-type excavation may require only high-pressure jets or chisels to facilitate material fracture. The upper Miocene marine deposits at the typical site are several hundreds of feet thick and should exhibit relatively uniform distribution both horizontally and vertically.

Underlying formations of older geologic time are of little importance.

The typical southern California coastal site may be underlain by the middle Pliocene San Mateo formation rather than the upper Miocene marine formation described above. This uniformly distributed, massive, well-graded, coarse-grained sand formation also exhibits high density and strength and extends vertically downward from elevation +20 for several hundred feet.

3.5 HYDROLOGY

The average annual rainfall along the entire coastline of southern California is in the 15-inch range. This rainfall occurs almost exclusively during the winter months of the year and almost always as storm rainfall. At a typical undeveloped site, runoff from rainfall on nearby hills and coastal plain will collect in ephemeral streams and discharge into the ocean at a number of locations along the beach. Generally, site drainage presents no difficulties and requires a minimum amount of culverts and channel improvements or construction.

Although all subsurface soils should be saturated, there should be no appreciable ground water storage above sea level at the typical site. Above sea level, any subsurface flow of water will be seaward, occurring primarily during the period immediately following heavy rainstorms. Below sea level all interstices should contain sea water, and subsurface flow can, and probably will, occur through crevices and natural fractures from the sea towards and into excavations. Ground water should move relatively slowly through the well-compacted San Mateo formation, however.

3.6 SEISMOLOGY

Next to Alaska, California experiences more seismic activity than any other state in the United States. The San Andreas fault zone traverses the State on a NNW to SSE general heading from Point Arena to the Mexican Border near El Centro. This fault zone is considered the major North American trace of the world's most active seismic belt, and it has been the source of many severe earthquakes. There are 15 other seismically active fault zones in California, 13 of them in southern California. These zones are shown in Figure 3-2. No coastal site in southern California is more than 40 miles from a seismically active fault zone, and most are within 25 miles. As a general indication of seismic activity in southern California, it is noteworthy that of the 2433 California earthquakes of Richter-scale magnitude 4.0 or greater during the period 1931 through 1964, 1470 were centered in southern California. Several of these earthquakes caused serious damage to man-made structures; one, the March 10, 1933, Long Beach earthquake, severely damaged structures along the coastline.

Seismic regionalization maps published in 1958 by Dr. C. F. Richter indicate maximum expected intensities of earthquake-induced ground motion along the southern California coast of from VII to IX on the modified Mercalli scale. Regionalization involves considerable judgment as to the influence of distance to active fault zones and specific site geology. It is therefore difficult to determine an accurate single value of the expected maximum earthquake intensity at a "typical" southern California site. A maximum intensity of VII, modified Mercalli appears reasonable from consideration of generalized site geology alone. Based upon the formula derived from Richter and Gutenberg relating intensity (I) and acceleration (a) as $\text{Log } (a/g) = (I/3) - 3.49$, the maximum ground acceleration expected at the typical southern California site will be approximately 0.07 g. Nevertheless, for those few areas along the southern California coast where seismically-induced ground motion of intensity IX can be expected, a ground acceleration of almost 0.33 g could reasonably be expected during a severe earthquake. Therefore, the design horizontal seismic load at a typical southern California coastal site will fall somewhere between 0.07 g and 0.33 g. Although geology alone would indicate

selecting a design factor near the lower end of this range, the importance of the installation under consideration, the relatively serious consequences of damage from seismic activity, and the current uncertainty about precise location dictate use of the highest reasonable design factor. Consequently, the factor of 0.33 is used in the present analysis. The main coolant system is designed to resist elastically a horizontal and vertical acceleration of 0.33 g. Other portions of the facility are designed on a less restrictive basis to take advantage of their capacity to absorb energy through plastic action.

3.7 WAVES

In summer, the prevailing sea-swell breaking at the shore has a significant trough-to-crest height (average of the highest one-third present) of 2 to 6 feet and a period of 12 to 16 seconds. In winter the waves are more variable with the significant height from 2 to 10 feet and the period of 8 to 18 seconds.

The design wave significant height (based on the tropical storm of September 1939) would probably be about 26 feet. The period associated with this height would be 14 seconds at a typical location.

3.8 TSUNAMIS OR SEISMIC SEA WAVES

The west coast of the United States has experienced many tsunamis but few are of local origin. Generally, severe damage from seismic sea waves is confined to a relatively small number of locations along the coastline where unique off-shore bottom conditions prevail and where a bay or harbor tends to amplify the tsunami. Damage is caused by flooding because of the high water resulting directly from the tsunami and by the action of normal wind-generated waves or surf, upon areas or structures not normally subject to this action. It is assumed that the typical site under consideration will be along open coastline, not in an embayment. Therefore, allowance need be made only for normal increase in sea level due to a tsunami. On this basis, it is believed that a plant location protected by a sea wall 20 feet high will be unaffected by seismic sea waves of distant origin. A survey of data concerning tsunamis along the California coastline which originated during the past two decades at distant epicenters indicates absolute maximum high water in the range of 9 feet to 13 feet above normal.(2)

No sizable tsunami with an epicenter in the continental borderland off southern California has occurred within the past century. However, in 1812 an earthquake in or near the channel islands produced a tsunami which (from carefully-gleaned sketchy information) appears to have had run-up height of 15 feet at Ventura, 20 to 25 feet at Santa Barbara and 50 feet at Gaviota. However, it is impossible to determine at this time, and impractical to guess, what run-up could occur at a typical southern California site.

3.9 CURRENTS

Littoral currents are typical for most sites along the southern California coast line. These currents, which result from waves breaking diagonally to the shore, may cause undesirable accumulations and decrements of sediments around structures which project seaward from the shoreline. The magnitude and significance of littoral currents must be investigated in detail for each specific site.

3.10 KELP

Extensive beds of kelp occur in the nearshore regions along the southern California coastline. These massive growths of seaweed will impose restrictions on the location of intake and discharge structures. An intake must be located in water of sufficient depth to prevent the growth of kelp and to minimize the possibility of floating kelp entering the intake. The discharge line must be located so that heated brine will not interfere with the growth of kelp and other marine life.

3.11 METEOROLOGY

The meteorology of the southern California coastal area is characterized by the regularity of the general movement of the surface winds, on-shore during the daytime and off-shore at night. These winds are the strongest and most regular during the summer months but also occur to a somewhat lesser degree during the winter. The on-shore winds are from approximately due west while the off-shore winds are from the northeast. Winds from the south and southeast are very infrequent, occurring only when storms move across the area, in which case there is considerable turbulence and mixing. Average wind speed is about seven mph. Calms and low wind speeds (0 to 4 mph) lasting longer than a few hours are very infrequent, particularly during the summer. However, during the daily shift of wind from on-shore to off-shore and back, periods of low wind speed lasting up to 2 or 3 hours often occur. An elevated inversion condition is prevalent all along the coast during the summer. However, usually there is considerable volume for diffusion below the inversion base so the atmospheric dispersion should be relatively good most of the time.

Section 4

SITE DEVELOPMENT

4.1 GENERAL

Construction of the plant at the "typical" southern California coastal site described in Section 3 requires extensive excavation to provide a nearly level site in the coastal bluff. For the 50-mgd reference plant, a gently sloping area with an ocean frontage of approximately 1000 feet and a depth of 500 feet is required. The average elevation of the site is 27 feet above sea level. A protective sea wall, 20 feet high, protects and stabilizes the shoreline at the site. An access road connects the facility to the nearest highway network. No other site access is provided. Tie-ins to two independent sources of electrical power are provided: one to a 220-kv line and one to a 33-kv line. Potable and fire protection water supply are provided from in-plant sources. An on-site waste water treatment and disposal system is provided.

4.2 EXCAVATION AND BACKFILL

Excavation for the complete plant requires removal of approximately 800,000 cubic yards of material. This leaves a site sloped from 15 to 39 feet above sea level. All bulk excavation will be done with conventional earth-moving equipment, such as tractor dozers, scrapers and motorized graders. A minimum of clearing and grubbing is required to clear away a light cover of grass and scrub growth. All topsoil removed would be re-used at the plant; no outside supply of this type fill is contemplated.

The prepared site consists of a series of pads at nine different elevations arranged in three-foot increments from 15 to 39 feet. These pads would be formed by proper control of final excavation rather than by backfilling to desired grade. The reactor plant will be located on the uppermost (39-foot) level of the site. The entire area at 39-foot elevation is brought to grade by excavation.

The top of the sea wall is 20 feet above sea level. Some backfill is required to provide a uniform slope from plant elevation to this wall. A fill of about 20,000 cubic yards of material cut from elsewhere at the site should be adequate.

Disposal of the large quantities of excess material excavated from the site will depend upon site conditions. Some of this material may be used to fill barrancas on adjacent lands to improve drainage.

4.3 SEA WALL

A reinforced concrete sea wall provides shore protection for the site. The foundation for this wall is five feet below sea level and the top of the wall is 20 feet above sea level.

4.4 PIT EXCAVATION

Two 100-foot deep pits are required; one of 15-foot inside diameter for the reactor and one of 42-foot inside diameter for brine heaters, pumps and piping. Approximately four feet of concrete (or equivalent density nuclear shielding) is required between the two pits. Consequently, the reference design is based upon using two-foot thick concrete linings for each pit, these liners meeting tangentially.

The water table at the typical site is essentially at sea level, and excavation below this level will be "in-the-wet". Caisson-type excavation has been selected as the reference design to provide the two pits. In plan, the caisson has one rounded end of 7.5-foot inside radius, for the reactor pit, and the other end of 21-foot inside radius, for the brine heater pit. The two centers of curvature are separated by 32.5 feet. The sides of the caisson are tangent to both curved sections. The caisson walls are concrete, 6 feet 9 inches thick; the bottom plug is eight feet thick. These dimensions are based on strength requirements, anticipated surface friction during sinking of 500 pounds per square foot, and weight to overcome buoyancy. A 1/4-inch thick steel liner is provided on the interior lateral and bottom surfaces of the caisson as a moisture seal.

The caisson would be sunk in a pre-drilled trench or slot reaching to the final bottom depth of the pit. This trench can be formed by drilling a series of overlapping, large-diameter holes which are filled with a liquid clay suspension. Drillers mud or bentonite should not be required; local material may be used instead. The caisson would be built above this trench and sunk into place by jetting in the pre-drilled trench and excavating the material in the space inside the caisson. The tendency for the blunt, heavier side of the caisson to sink more rapidly than the rest of the caisson can be minimized by careful control of jetting and excavation. To minimize expensive caisson-type construction, and to include extra excavation to form a construction floor for caisson construction and sinking, excavation down to the water table would be accomplished by conventional means. This extra excavation would be backfilled after the caisson is complete.

When the caisson reaches final elevation, a tremie seal would be poured in the bottom and all water pumped from inside the caisson. The bottom plug and steel liner would be placed, and the concrete liners for the

two pits completed. The steel liners are fastened to the caisson by studs embedded into the concrete. The liner and studs would be installed as the caisson is built, or alternately the liners would be welded to studs which were placed in the concrete during caisson construction.

The caisson method of constructing deep pits should be generally applicable to any site condition along the southern California coastline, including sites with deep alluvium and sand strata. Nevertheless, the geology of the reference site is such that a mining-type operation also may be practical and in some cases may be more economical. The applicability of this type of construction will depend upon two factors: the capability of the walls of the excavation to stand with relatively little support, and economical control of ground water so that the excavation is essentially "in-the-dry".

The shaft opened by mining operations would be similar in plan to the pit constructed with a caisson. The mining operation would proceed as follows:

1. Shot holes would be pre-drilled to the ultimate pit elevation and backfilled with sand.
2. These holes would be cleaned out as required (in about 10-foot lifts), charged, tamped and shot.
3. Fractured material would be removed by clam shell.
4. A gunnite liner would be applied to the excavation walls to prevent rock fall, etc.
5. Forms would be set and the concrete liners for both pits poured, after the shaft has reached final excavation. The inner form could be the steel water-proofing liner for each pit.

The feasibility of the mining operation cannot be determined on a generalized site basis. When a final site is selected, test boring should be made to provide information necessary for proper evaluation of the cost of this construction technique.

4.5 ROADS AND PARKING

A two-lane access road connects the plant site to the nearest highway network. Parking hardstands are provided on site for large trailer trucks, light trucks and autos.

Section 5

REACTOR PLANT

5.1 GENERAL

Although this report describes and presents estimated costs of the entire water desalting plant, the study was directed primarily toward the conceptual design of the reactor plant and most of the engineering effort was spent in this area. This section describes the principal features of the deep-pool reactor plant conceptual design developed during the study. It also indicates some of the alternatives considered during the study and in some cases indicates the reasons on which the selection of a given alternative was based. For the most part, however, the conceptual design is based on common design practice and is intended to be similar to other reactor plants as far as possible. It is considered to represent a reasonable design approach and to be adequate for the purpose of developing a realistic cost estimate but it should not be construed as representing the optimum or necessarily recommended design for a plant of this type. Time was not available to consider all promising alternatives during the study. Further study and detailed design are likely to indicate cases where other approaches would be found to be superior to those selected.

5.2 REACTOR

5.2.1 Core

The primary emphasis of this study was placed on the reactor plant exclusive of the core to determine the capital costs and safety/siting features of a plant which could be built with essentially existing technology. The well-developed water reactor technology available for core design and fuel cycle costs de-emphasizes the need at this point for a detailed core design, recognizing it would be required before construction of a reactor of this type.

In this context, the basic core design presented is intended to be representative of what could be achieved with existing technology, rather than the result of extensive calculations and optimization for the conditions pertinent to this reactor.

The basic design results from a review of published and proprietary water reactor designs of comparable power, and scoping calculations by Chicago Operations Office.

The general criteria for the core of the 50-mgd plant are as follows:

1. Power level - 400 Mwt
2. Coolant - demineralized light water
3. Coolant temperature in - 200°F
4. Coolant temperature out - 270°F
5. Flow direction - down
6. No net steam generation
7. Lowest practical pressure drop core design
8. No fuel hold-down structure
9. Sufficient movable or fixed poison for control and shutdown
10. Inherent shutdown and/or power limiting capability

Selection of Materials

Fuel Materials and Form: The fuel materials which might reasonably be considered established technology for this low-temperature low-pressure reactor would be:

1. Low enriched UO_2
2. $\text{UO}_2\text{-ThO}_2$
3. Metal alloys of uranium

Low enriched UO_2 was selected for the fuel material because it has been widely used, and technical and economic information is readily available. The fuel form was assumed to be pellets in cylindrical rods, no analysis of the merits of plate-type cores having been performed.

While no economic analysis was done to establish fuel cycle costs for the various fuel materials and forms, the choice of low enriched UO_2 should result in fuel cycle costs representative of those which can be achieved with one or more of the other fuel materials.

Cladding Material: The cladding materials which were considered because of their established technology are:

- LEGEND**
- Qt Terrace Material: brown clayey sand, silt & gravel.
 - Mu Upper Miocene Marine: siltstone, sandstone, shale.
 - Qbs Beach Sand: fine & medium clean sand.

TYPICAL GEOLOGIC CROSS SECTION
SOUTHERN CALIFORNIA COASTAL BLUFF

FIGURE 3-1

FIGURE 3-2

1. Aluminum alloys
2. 300 series stainless steels
3. Zirconium alloys

Aluminum alloys were not chosen because no significant quantity of UO_2 clad with aluminum alloy tubing has been irradiated under environmental conditions or to burnups of interest to this reactor. In addition, the desire to use low cost copper alloys for brine heater tubing and the potential for chlorides in the primary water could lead to pitting of the aluminum cladding from copper and chloride ions in the primary system water.

In view of the recent experience with stainless steels in several boiling water reactors, and without a detailed evaluation of water chemistry and materials properties, the potential requirement for temporary compatibility with high chloride content coolant was assumed to eliminate the stainless steels from consideration.

Zircaloy-2 was chosen as the reference clad material because:

1. It has performed successfully under high fuel burnup conditions, and under flow, temperature, and pressure conditions comparable to and substantially higher than those for this reactor.
2. The fuel fabrication costs with this cladding are known with reasonable accuracy and the cost of Zircaloy components is decreasing.
3. It is comparable to aluminum alloys from the standpoint of neutron economy.
4. Its integrity in sea water is understood to be excellent in the unirradiated condition. The Zircaloy-2 clad UO_2 in the Vallecitos Boiling Water Reactor performed satisfactorily for at least short periods of time with high chloride content in the coolant, with no known deleterious effects.
5. The low temperature ductility of Zircaloy-2 has been shown to be adequate in four-year Hanford production reactor tests ⁽⁴⁾ of Zircaloy pressure tubes at test temperatures ranging from 20°C to 300°C.

Control Rod Materials: From the variety of materials and forms in use or under development, B_4C powder in Inconel tubes has been arbitrarily selected as the basic control material. The small diameter B_4C -filled

tubes are assembled in a cruciform structure and sheathed with a thin, perforated sheet of Inconel.

Rods of this type are light weight, relatively low in cost, and are being or will be used in numerous boiling water reactors including Humboldt Bay, Big Rock Point, Dresden, LaCrosse, Elk River, Oyster Creek, and Nine-Mile Point.

Burnable Poisons: To provide sufficient control and shutdown margin it has been the practice to include sufficient movable rods. This approach, while obviously adequate, is expensive. To reduce the number of movable rods, a variety of alternatives are now or will be used in several power reactors. Examples include:

1. Fixed sheets or shims to limit initial excess reactivity. Examples include removable fuel channels (stainless replaced with zirconium) at Dresden, replaceable poison pins in the fuel assembly at Elk River, burnable poison strips at EBWR, and shims in the Pathfinder Reactor.
2. Burnable poison in the fuel or fuel cladding. Examples include borated cladding at Elk River and Indian Point, and UO_2 containing 0.35 per cent Er_2O_3 at Dresden.
3. Soluble poison to provide adequate shutdown capability and/or compensate for the reactivity shift as fuel is consumed. Examples include the EBWR and Yankee Reactors.
4. Limiting the initial size of the core until self-generated burnable poisons (Xe and Sm) are present.

It is assumed that the core contains burnable poison or neutron absorbing material in one or more of the above forms.

General Description of the Reactor

The preceding paragraphs have described the general criteria for and selection of several of the major features for the core. The following summarizes the description of a core which is believed representative of what could be achieved with present technology.

1. General

Reactor power - 400 Mwt
Fuel - Zircaloy-2 clad UO_2
Coolant flow rate - 40,000 gpm

2. Heat Transfer

Power density - 50 kw/liter of core
Heat flux (average) - 100,000 Btu/hr-ft²
Heat flux (maximum) - 325,000 Btu/hr-ft²
Critical and burnout heat flux $>1.0 \times 10^6$ Btu/hr-ft²
Coolant inlet temperature - 200°F
Coolant outlet temperature - 270°F
Pressure at core outlet - 77 psia
Saturation temperature at core outlet - 310°F
Fuel temperature (overpower) $<5000^\circ\text{F}$

3. Core

Number of fuel assemblies - 120
Number of control rods - 30, on a 12-in. pitch (no follower)
Active height - 6-1/2 ft
Active diameter (equivalent) - 6-1/2 ft
Fuel assembly shape - square, 5-7/8 x 5-7/8 in.
Fuel rods enclosed in square channel
Overall fuel assembly length - 8 ft

4. Nuclear Parameters (equilibrium)

Fuel loading - 25,000 kgU
Initial enrichment - 2.75%
Final enrichment - 1.15%
Average conversion ratio - 0.55
Discharge burnup - 20,000 mwd/t
(22,000 mwd/mt)

The inherent shutdown/power limiting features of a water reactor core are available. The use of UO₂ provides a negative Doppler coefficient. A relatively tight fuel rod lattice would provide negative temperature and void coefficients and a satisfactory conversion ratio.

The discharge fuel burnup of 20,000 mwd/t (22,000 mwd/mt) results from an annual refueling of approximately one quarter of the full core loading with a 90 percent plant factor.

Thermal-Hydraulics

The 400 Mwt of heat generated in the core is removed by 40,000 gpm of demineralized water flowing down through the core. The water enters the core at 200°F and exits at 270°F. The gridplate and/or fuel assemblies are orificed to provide approximately 270°F exit temperature from each fuel assembly. A removable channel around each fuel assembly

provides positive flow distribution and permits (through the use of different channel materials) adjustment of initial reactivity and core power distribution.

The flow baffle extending downward from the core contains appropriately sized cutouts to provide good flow distribution at the core outlet and to the reactor tank outlet nozzle.

The 40,000 gpm rate includes an allowance for bypass flow (e. g. , around control rods). The neutron/gamma shielding in the annulus between the core edge and reactor tank is cooled by natural convection. A plate in the annulus between the core and the tank forces the coolant to flow through the core.

Steady State Operation: Scoping calculations indicate that there will be no net steam produced by the core with approximately 40°F subcooling at the core exit (270°F versus 310°F). With orificed fuel channels, 270°F bulk outlet temperature, a peak heat flux of approximately 325,000 Btu/hr-ft², a film coefficient of approximately 3000 Btu/fr-ft² °F, and use of existing technology to obtain a relatively flat power distribution and reduce hot spots, no nucleate boiling would be expected. The peak possible clad temperature would be determined by the cladding surface temperature above saturation required to initiate nucleate boiling. Reference (5) and communication with the author indicates that a peak surface temperature in excess of approximately 350°F (40°F above saturation) is required to provide sufficient superheat to initiate nucleate boiling.

Methods available to flatten core power distribution and reduce hot spots include (1) lower enrichment, solid, burnable poison, or thicker clad corner fuel assembly rods, (2) burnable poison in the coolant, fuel, cladding, or in strips or fuel channels, (3) initial and subsequent fueling methods, (4) control rod programming, (5) control rod followers, and (6) soluble poison control.

Pressure drop calculations for several fuel rod spacings (water-to-fuel ratios from 1.0 to 1.6), and actual pressure drop data for several existing fuel and hardware geometries indicate that with a channel coolant velocity of 8 to 9 ft/sec and careful hydraulic design, a core pressure drop of approximately 5 psi can be obtained.

The calculated critical and burnout heat fluxes are in excess of 1×10^6 Btu/hr-ft² (6,7).

The desire to eliminate boiling was established arbitrarily. Upon further analysis, some boiling might be permitted.

Shutdown Transient: The use of coolant flowing down through the core has been demonstrated in numerous small pool or tank type research reactors. The Piqua Reactor, the high power density Belgian BR-2 test reactor, the Materials Test Reactor (MTR), and the Engineering Test Reactor (ETR) also use down flow.

Under normal shutdown conditions, flow through the brine heaters would be maintained for some minutes to reduce decay heat to an acceptable level.

Under emergency conditions in which power to the main coolant pumps is lost, the reactor would be scrammed and the flywheel or built-in motor inertia on the two half-capacity pumps would provide several minutes of flow through the core. One example of the use of flywheels is the PRTR reactor in which flow coast-down times in the range of 3 to 5 minutes have been obtained⁽⁸⁾.

For the case where primary flow is somehow stopped at the time a scram occurs, calculations indicate the residual specific and decay heat would be removed by short term boiling of the coolant with flow up the center fuel assemblies and down the peripheral assemblies, the core acting as a natural circulation loop. Though apparently not required, a check valve(s), or equivalent device(s), could be provided in the baffle plate in the annulus outside the core, to open on the loss of primary flow. In this case the flow would be up through the core and down the annulus outside the core. This approach has been used in pool-type research reactors with forced-circulation downflow systems.

The ability to sustain (without damage) a flow stoppage immediately after a scram has been demonstrated in monitored tests in the Belgian BR-2 Test Reactor. The results of these flow reversal experiments are reported in Reference⁽⁹⁾.

Core Structure

The core structure consists of the core support flange, outlet flow baffle, lower grid or fuel support plate, upper grid or fuel guide plate, control rod guide shroud, and other locating, bracing, and flow baffle and supporting members.

The weight of the fuel and core structure is supported from a circumferential flange on the inside diameter of the reactor tank. The neutron and gamma shields are also supported from this flange.

The fuel assemblies are located and supported by the lower grid plate which consists of a flat plate with supporting webs attached to the lower surface of the plate. The resulting structure is analogous to a plate consisting of T-beams.

The control rod guide is attached to the upper guide plate. The control rods are guided in the active core by the walls of the channel on each fuel assembly.

A flow baffle plate and sway braces are located in the annulus outside of the core to eliminate undesired core flow bypass and provide lateral support for the core structure.

The core structure described does not provide for control rod followers. If followers were to be used to help prevent nucleate boiling on the hot pin(s), space is available beneath the lower grid plate for a follower guide structure.

Shielding

The shielding is the neutron/gamma shielding required to eliminate concern regarding the need for any cooling in the concrete pit wall.

The shielding, which is located in the annulus between the core edge and reactor tank wall, consists of approximately 18 inches of water, one inch of boronated carbon steel, and three inches of lead canned in carbon steel. The materials and thicknesses represent the results of a review of existing shielding installed in the EBWR, Elk River, and LaCrosse reactors.

The boronated steel and canned lead cylinders are located in the reactor tank to eliminate the requirement for a separate cooling system(s). The low coolant temperature permits the use of canned lead in the tank.

The practicality of fabricating and natural convection cooling of the canned-lead gamma shield has been confirmed by an informal discussion with a manufacturer of spent fuel shipping casks and a review of the heat transfer characteristics of several existing canned-lead, spent fuel shipping casks. The Elk River-Piqua cask is of particular interest in that it is a dry cask with no internal or external forced convection cooling system.

Boronated carbon steel has been fabricated and used for neutron absorption purposes in at least two locations, one for control rods (Brookhaven) and the other as part of the NPR reactor biological shielding (Hanford).

It is believed that detailed shielding calculations would permit a reduction in the 18-inch water thickness. This would permit a slight reduction in the diameter of the reactor tank.

5. 2. 2 Control Rod Drives

Criteria

In selecting a drive, several desirable criteria were established. Among these are that the drive:

1. Be based on established technology to the maximum extent practical.
2. Be simple and capable of reliable operation in its environment.
3. Be low in cost and not require costly building space or auxiliaries.
4. Be readily accessible for inspection, maintenance, or replacement.
5. Insert, withdraw, and hold control rods in the same manner being used in comparable power reactors.
6. Provide a positive insertion force in addition to the weight of the control rod itself.
7. Provide for rod position indication.
8. Not unnecessarily extend refueling time or interfere with refueling operations.
9. Minimize penetration of reactor tank.

Drive Location

Five basic drive locations were considered. They are:

1. Bottom-mounted - below the reactor tank
2. In the tank - below the core
3. In the tank - just above the core
4. Side-mounted - outside the tank just above or below the core
5. Top-mounted - in the air above the tank water level

Based on the general criteria noted above and on discussions with three reactor manufacturers and one control rod drive manufacturer, top-mounted drives were selected.

The advantages of top-mounting include:

1. The drive is readily accessible for maintenance, repair, or replacement.
2. There are no design requirements for pressure shells or seals, underwater bearing surfaces, suspended crud, flushing, cooling, etc.
3. The accessibility, air environment, and relatively simple design requirements result in a low cost drive.
4. There are no penetrations of the reactor tank for control rod drive shafts and no electrical leads in the water.
5. There is no concern about flooding the region in which the drives are located in the event of system or ground-water leaks.

Drive Types Considered

The decision to base the design on established technology led to consideration of such types as rack and pinion, magnetic jack, hydraulic, and cable-type control rod drives. Each of these types is in use today and is entirely feasible for use as the basic drive; the costs in an air environment are about the same. For the reasons noted in the next section, a magnetic jack was selected as the drive type for this study.

Reference Design

The magnetic jacks and dashpots are mounted on the reactor tank lid. The drive rod which passes through the magnetic field is connected to a 130-foot long water-filled extension pipe, connected in turn to the control rod itself. The control rod remains connected to the extension pipe under all operating conditions.

The extension pipes are aligned and guided by a retractable guide structure attached to the reactor tank above the water level. A similar guide structure supports the extension pipes laterally in the reactor tank just above the tank inlet. Following disconnect from the rods and drives during a refueling shutdown, the pipes are supported from a notched flange at the top of the reactor tank.

Position indication can be provided by one of several different types of sensors mounted on the jack.

For refueling operations, each drive rod is disconnected from its associated extension pipe. The drive package is then removed, leaving

each extension pipe supported by the framework. The control rod latch is actuated, and the extension pipe picked up slightly and placed in its support bracket at the side of the tank, leaving the control rod in the core. The extension pipe is not normally removed from the tank. The end result is a completely open cylinder of water down to the core.

Under scram conditions, the weight of the extension pipe can provide an added force to the control rod. This force would be in the 300 to 500 pound range depending upon the pipe diameter and thickness.

A standard three to four-inch pipe approximately 130 feet long will not buckle under pinned-end conditions. Though it is not normally required since the pipe is in tension, a pipe of this size is useful for delatching and forcing the rod down by applying a compressive load.

As with most other drives, a proof test under simulated operating conditions would be required to demonstrate the practicality of the proposed drive arrangement and establish final design requirements.

It is recognized that other top-mounted drive types could be used. The basic reasons for selecting the magnetic jack type are:

1. Its cost for this application is comparable to other types.
2. It has the least number of moving parts, with a potential for increased reliability.
3. The drive package can be easily disconnected from the extension pipes.
4. It requires no lubrication and a minimum of preventive maintenance.
5. It is totally enclosed.

5.2.3 Nuclear Instrumentation

Conventional reactor control and safety instruments would be used with both local and control room mounted components such as ion chambers, power supplies, indicators, log count and period meters, recorders, preamplifiers, etc.

The basic signals are provided by the required number of cable-hung fission and ion chambers located in the air-filled annulus between the reactor tank and concrete pit wall. This location permits relatively simple chamber positioning and removal for replacement. The use of this lower temperature air environment eliminates the need for reactor tank penetrations and chamber waterproofing.

The background count neutron source could be located in a dummy fuel assembly or attached to a cable and positioned in a tube at the outer edge of the core. The latter approach has been selected.

5.2.4 Refueling

In selecting a refueling method, the general requirements established were that the approach:

1. Be based on complete underwater transfer of spent fuel from the core to the storage rack to avoid the need for a transfer cask and accessories, the potential for dropping a cask in the tank, and the possibility of an operating floor fuel drop accident with its associated containment requirements, e. g. , Humboldt Bay Reactor.
2. Be simple, direct and low cost, with the refueling equipment accessible for maintenance, repair, or replacement.
3. Not unnecessarily extend refueling time.
4. Not require costly building space or auxiliaries which are used but once a year.
5. Provide for spent fuel storage in a conventional pool near grade level to reduce excavation costs, and provide 100 percent accessibility for removal of spent fuel from the reactor building without interfering with reactor operation.
6. Be based on direct visual observation of the refueling by the operator.
7. Minimize exposure of the carbon steel tank to air.

Items 1, 5, and 7 eliminate those methods which would involve lowering the water level to some 20 or 30 feet above the core, and effectively require the operations to be performed or directed from the top of the tank.

Items 2 and 4 suggest maximizing the use of human skills rather than using complex programmed or indexed equipment subject to malfunction.

Item 6 suggests the use of removable underwater lights and television camera(s) to get the operator's eye close to the top of the core.

Calculations indicate that at the time refueling operations could begin (one day assumed) that the coolant velocity, under the natural circula-

tion conditions then existing, would not exceed approximately one-half ft/sec. As a result, hydraulic forces will be very low.

Based on the above, the approach selected for refueling consists of a refueling bridge at the top of the tank, an underwater light(s) and television camera(s) mounted on a cable(s) with the television monitor or screen located on the bridge. The grapple consists of a pipe with a latching device at the bottom. A special removable guide plate can be placed at the top of the core structure if required. A simple basket and cable are provided at the side of the tank to raise the fuel assembly from the core region to the transfer chute level for transfer to the spent fuel pool. The use of the basket permits grappling and removal of the fuel from the core while an assembly is being raised and transferred to the spent fuel pool, thus reducing refueling time.

The selected method is illustrated in Drawing EA5.

Refueling alternatives considered but not selected include (1) a removable in-tank manipulator with a television system, and (2) raising the entire core to the upper part of the tank for direct manual refueling.

5.3 MAIN COOLANT SYSTEM

5.3.1 General Description

The main coolant system consists of a reactor, reactor tank, coolant pumps, brine heaters, and associated valves and piping. Auxiliary or peripheral systems which accomplish the various functions such as waste disposal, filtration and purification, ventilation cooling and circulation, and auxiliary cooling are discussed in the following section of this report. The main coolant and auxiliary systems are shown schematically in Drawing EA6.

High-purity coolant is circulated in the main coolant system at a flow rate of 40,000 gpm by the two coolant pumps. Flow is downward through the core to a discharge plenum, and then to a single outlet nozzle in the side of the reactor tank approximately four feet above the bottom. Coolant temperature at the tank outlet is 270°F. From the tank, the water flows to the suction of the pumps, and from the pumps to a header and then to the first two brine heaters, which operate in parallel. The divided flow merges at the exits of these heaters, and then passes successively through the third and fourth heaters, emerging from the latter at a temperature of 200°F. It then re-enters the reactor tank through a single nozzle about 40 feet above the tank bottom. The reactor tank is 150 feet in height and 11 feet in diameter. Shutoff valves are located at

the reactor tank inlet and outlet and check valves are located at each pump discharge. A valved bypass line is connected from the pump discharge header to the inlet of the third stage brine heater. A coolant overflow line is located approximately three feet below the top of the reactor tank. This two-inch line feeds into the liquid radwaste collection tank. The total volume of the primary coolant system is approximately 290,000 gallons.

All major components of the main coolant system are carbon steel except for the reactor, brine heater tubes and tube sheets, and pump and valve trim. The principal components of the main coolant system are discussed in the following paragraphs:

5.3.2 Reactor Tank

Diameter - 11 ft
Length - 150 ft
Volume - 110,000 gal.
Design pressure at bottom - 100 psig
Wall thickness - 3/8 in.

The reactor tank is fabricated from carbon steel, SA 516, grade 70. The tank is supported vertically by a skirt at the bottom; this skirt is 3/8-inch thick and rests on a lipped metal foundation plate. The skirt "foot" is lubricated by a lubrite pad arranged so that the vessel is self-centering and free to move as a result of thermal expansion. Lateral supports (seismic anchors) are provided on the vessel at elevations -100, -60, 0 and +50 feet. The vessel is free to move radially and vertically in these supports.

A flat tank top is sealed to the tank by means of a light-weight flanged and gasketed joint. The housings for the magnetic jack control rod drives are connected directly to the tank top. A refueling slot is provided in the side of the tank. This is a rectangular nozzle approximately 1.5 feet wide and 17 feet high, terminating at the top of the tank. The nozzle is flanged and normally covered with a plate which is bolted and gasketed to the nozzle flange. For refueling, the cover plate is removed and a refueling trough is installed.

The bottom head of the reactor tank is torispherical. This type of head minimizes material requirements and stress concentrations at the head to cylinder junction. Additionally, the skirt-type support readily accommodates thermal expansions and contractions. A flat bottom head might simplify reactor support but would tend to create unsatisfactory stress concentrations and would not as readily accommodate thermal movements.

5.3.3 Brine Heaters

1. First Stage Heaters

Quantity - 2

Tube Length - 64.25 ft

Shell OD - 10.8 ft

Volume of tubes (brine) - 11,500 gal.

2. Second and Third Stage Heaters

Quantity - 1 each stage

Tube Length - 70.6 ft

Shell OD - 12 ft

Volume of shell side (reactor coolant) - 45,700 gal.

Volume of tubes (brine) - 12,000 gal.

Brine heater design data are included in the description of the water plant in paragraph 6.3 and in Table 6-2. The reactor coolant flows successively through the three stages of brine heaters. The first stage consists of two heaters and the last two each have a single heater. Reactor coolant flows in the shell and brine flows in the tubes. The four heaters are arranged vertically in the brine heater pit and are completely below grade. The heaters are supported on skirts.

The shells are fabricated from carbon steel to the same specifications as the reactor tank. Differential thermal expansion between the tube sheets and shell must be accommodated and it has been assumed that either some form of shell expansion joint would be provided or bent tubes would be used.

The brine heater tubes are 70-30 cupro-nickel and are rolled into double tubesheets. The tubesheets are either solid cupro-nickel or cupro-nickel-clad carbon steel. The space between the tubesheets is connected to a leak collection/detection system.

The shell and tube sides of the brine heaters can be separately drained and the drainage pumped to the large waste collection tanks in the rad-waste system. In the event of leakage which might contaminate either stream (the brine is normally maintained at a pressure slightly higher than the reactor coolant), the brine line to the leaking heater would be closed and the heater drained and refilled with clean water, after which the plant could be operated at reduced capacity until shut down for tube plugging.

5.3.4 Main Coolant Pumps

Quantity - 2
Type - Centrifugal
Capacity - 20,000 gpm
Power - 700 hp
Speed - 900 rpm

The pumps provide a net pressure rise of 115 feet of water and are of the limited-leakage type having mechanical shaft seals. Because of the low operating pressure of the pumps (approximately 65 psig), the shaft seals should be extremely reliable and have almost negligible leakage.

All of the pump components are carbon steel, except the shaft, wear rings, etc., which are stainless steel. The pumps are mounted on the piping but have lateral sway struts which tie back to the brine heater pit wall. Coolant enters the pumps from the bottom and is discharged from the side.

The pump motors are equipped with fly wheels or extra inertia is built into the motor rotor to provide a slow flow coastdown in the event of a loss of pumping power.

5.3.5 Valves

1. Main Coolant Stop Valves

Quantity - 2
Type - Gate
Diameter - 40 in.

2. Check Valves

Quantity - 2
Type - Stop-check
Diameter - 30 in.

3. Main Coolant Bypass Valve

Quantity - 1
Type - Butterfly
Diameter - 30 in.

The main coolant stop valves are motor-operated 40-inch gate type. The motor operators are designed to prevent rapid closure of these valves and initiation of the closure causes the reactor to scram. The coolant bypass valve is a metal-seated butterfly type with an operator designed to open in the event of loss of power. The pump check valves are 30-inch valves. All valves will have carbon steel bodies and stainless steel seats and miscellaneous trim.

5.3.6 Piping

Piping is all welded carbon steel. Reactor inlet and outlet lines and all lines normally taking the total flow are 40-inch diameter; the pump inlets and outlets and all lines taking half the flow are 30 inch. Welding will be used wherever possible and flanges used only where essential for component removal.

5.4 REACTOR AUXILIARY SYSTEMS

5.4.1 General

The reactor auxiliary systems consist of the Coolant Purification System, Deaeration System, Radioactive Waste Disposal System, Spent Fuel Pit Cooling and Purification System, Decay Heat Removal System, Component Cooling System, Emergency Poison Injection System, Ventilation and Air Cooling System and the Sea Water Cooling System. All major components of the systems are carbon steel unless otherwise noted.

Flow diagrams for each system are included in the text; Drawing EA6 shows all the auxiliary systems together with the Main Coolant System.

5.4.2 Coolant Purification System

The function of the coolant purification system is to remove impurities from the coolant to limit the level of induced radioactivity in the coolant, and to minimize deposition of radioactive crud and the fouling of heat transfer surfaces. The principal sources of impurities are corrosion products from the brine heaters, tank wall, and pipe surfaces. The system design is based on maintaining the level of these impurities in the coolant below one ppm.

System Description

Reactor coolant is taken from the entrance to the second brine heater at a rate of 500 gpm. Sufficient head exists at this point in the loop to furnish the required flow through the purification system. The water passes first through a regenerative heat exchanger which reduces the coolant temperature to 150°F. A non-regenerative exchanger then brings the temperature to 120°F. After passing through the heat exchangers, the water flows through an ion exchanger and filter. Finally, after passing back through the regenerative heat exchanger, the water is returned to the main coolant system. The flow diagram for this system is shown on Figure 5-1.

Provision is made for passing water from other sources through the ion exchanger. These other sources include make-up water and low level activity liquid from the radwaste system. The de-ionized waste liquid may then be fed either to the main coolant system or back to the liquid radwaste system for disposal.

When the reactor and main coolant pumps are not in operation, the coolant may continue to be passed through the purification system by making use of a bleed-off from the decay heat removal circuit.

Component Description

1. Regenerative Heat Exchanger

Quantity - 1

Design flow rate - 500 gpm

Design heat load - 20×10^6 Btu/hr

2. Non-regenerative Heat Exchanger

Quantity - 1

Design flow rate (reactor coolant) - 500 gpm

Design flow rate (sea water) - 375 gpm

Design heat load - 7.5×10^6 Btu/hr

FLOW DIAGRAM
COOLANT PURIFICATION SYSTEM

FIGURE 5-1

3. Purification System Ion Exchangers

Quantity - 2

Volume of Resin Bed - 155 ft³

4. Purification System Filters

Quantity - 2

Type - Cartridge

Design flow rate - 250 gpm

Ion Exchangers: The rate of corrosion product removal is estimated to be 1500 pounds per year. This is based on estimated yearly corrosion rates of 0.5 mil/year for all carbon steel surfaces and 0.1 mil/year for the 70-30 cupro-nickel surfaces of the brine heaters. Two seven-foot diameter mixed resin beds are provided to remove this amount of corrosion product. Each bed is four feet deep and designed to last for six months. A typical resin would be one of the mixed type such as Amberlite IRC-150. To maximize the resin life, the maximum temperature of the inlet feed is designed to be 120°F. Spent resin will be sluiced to the resin disposal tank, and fresh resin added from the resin storage tank.

Regenerative Heat Exchanger: Water enters the hot side of the regenerative heat exchanger at approximately 230°F and leaves at 150°F. At the design flow rate of 500 gpm, this amounts to a heat duty of 20×10^6 Btu/hour. The return water will enter at 120°F and leave at 200°F. The non-regenerative exchanger cools the primary water to 120°F by removing 7.5×10^6 Btu/hour.

5.4.3 Deaeration System

A deaeration system is provided to remove dissolved air, hydrogen and oxygen. It is designed to operate while at power or while the reactor is shut down.

System Description

A simple spray nozzle pipe is located within the reactor tank, above the coolant. During operation, this pipe is supplied by a tap from the feed line to the coolant purification system. Since this water is at a temperature of approximately 250°F, spraying into the top of the reactor tank will cause a portion of it to flash. The flow diagram for this system is shown in Figure 5-2.

To deaerate during the post-operative cooldown period, feed to the spray pipe is supplied by a tap from the decay heat removal pump outlet.

Gases and water vapor in the top of the tank are bled off through a condenser, where the bulk of the water vapor is condensed. The non-condensable gases are then conducted to the gaseous radwaste system. If recombination does not take place in the reactor coolant, a re-combiner may be required, either before or after the condenser, to recombine hydrogen and oxygen formed by radiolytic decomposition of the water.

Component Description

With the exception of the spray nozzle and pipe described above, the major components in this system are parts of other systems and are described with those systems.

5.4.4 Radioactive Waste Disposal System

All plant wastes which are normally or potentially contaminated with radioactive materials must be controlled and disposed of in an appropriate manner to assure that the hazard to personnel, plant, and environment is minimized. These paragraphs describe the equipment and facilities provided for this purpose.

Liquid Waste System Description

The system and equipment for collection, segregation, treatment, and disposal of liquid wastes are shown on Figure 5-3.

* HEAD SUPPLIED BY MAIN
COOLANT PUMPS.

FLOW DIAGRAM
COOLANT DEAERATION SYSTEM

FIGURE 5-2

Waste liquid from the reactor and brine heater pit drains, pump leakages, and other miscellaneous overflows and drains are collected in the main liquid collection tank and from there pumped to one of the liquid waste tanks. Four of these tanks are available, two of 100,000-gallon and two of 10,000-gallon capacity. During normal operation, both of the smaller tanks are available for liquid radwaste processing. The large tanks are held in reserve for emergency storage from either the brine heaters, the spent fuel pit, or the reactor tank.

A 50-gpm centrifugal pump is provided for transfer or recirculation of the contents of the tanks and will also feed the radwaste demineralizer.

From the circulation pump, piping is provided to permit:

1. Recirculation to any of the four tanks.
2. Feed to the radwaste demineralizer.
3. Discharge to the ocean or to the drumming facility.
4. Return to the spent fuel pit.
5. Feed to the primary system purification demineralizer.

Laboratory radioanalysis of the collected wastes will be carried out to establish the treatment required and disposal limitations on individual waste batches. Aqueous wastes gradually accumulate in one of the 10,000-gallon liquid waste tanks, with the level indicated on the main control board. Collected waste in a given tank would be recirculated to ensure thorough mixing before sampling for analysis. A radio-assay for specific isotopes which might limit discharge of the wastes would establish the degree of treatment required for individual batches. The waste may then be discharged or demineralized as required. Treated wastes would be routed to an available waste tank for storage, pending analysis to confirm their suitability for discharge.

Provision is made for routing liquid wastes to the main coolant system purification demineralizer in the event that additional cleanup capability is required.

Pipe and valves are generally exposed (not shielded) for maintenance and manual operation unless they normally would be expected to carry significant activity levels.

Provision is made for the collection of large quantities of liquid arising out of abnormal or emergency situations. The 100,000-gallon liquid waste tanks are kept available for such situations. A possible

occurrence is the rupturing of a brine heater tube. In this case, the brine side of the heater would be isolated, drained, and flushed, with the liquid being pumped to one of the large tanks by the brine drain pump. It should be possible to discharge this liquid directly to the ocean from the waste tank.

After plant shutdown, the faulty heater can be drained, along with its neighboring heater, and pumped to one of the large liquid waste tanks for storage. This liquid can at a later time either be returned to the main coolant system or discharged to the ocean.

Liquid Waste Component Description

1. Pumps

<u>Function</u>	<u>Power (hp)</u>	<u>Flow (gpm)</u>
Main collection pump	2.5	50
Circulation pump	2.0	50
Sump pumps (2)	1.0	20
Brine drain pump	50	1000
Coolant auxiliary pump	75	1500

2. Tanks

<u>Function</u>	<u>Capacity (gal.)</u>
Liquid waste tanks (2)	10,000
Liquid waste tanks (2)	100,000
Resin disposal tank	10,000
Fresh resin tank	2,250
Liquid collection header	500

3. Radwaste Demineralizer

Quantity - 1
Resin bed volume - 13 ft³

Radwaste System Pumps: All pumps have mechanical shaft seals to minimize leakage. Materials of construction are conventional all-iron with steel shafts and stainless steel shaft sleeves.

Radwaste System Tanks: All radwaste treatment storage tanks are vertical, right-cylindrical tanks with self-supporting cone roofs, of fusion-welded carbon steel construction in accordance with API Standard 650. Manholes are provided for inspection access. A pump could be installed (in the future) on the manhole of the resin disposal tank should it become desirable to empty it.

The liquid radwaste tanks are provided with a baked phenolic lining. The resin disposal tank is also provided with a baked phenolic lining to prevent possible corrosion during long-term storage of spent ion-exchange resins.

Radwaste Demineralizer: A mixed resin bed with a two-foot diameter and four-foot height, designed for a normal throughput of 20 gpm, is provided for removal of dissolved contaminants in the liquid wastes.

Gaseous Waste System Description

Waste gases are pumped to the main header from two sources: bleed-off from the main liquid radwaste header, and bleed-off from the reactor tank collection space. This gas is compressed to a gas surge drum, and a bleedback system is employed in order to maintain a constant pressure on the waste gas header. Initially, this system is filled with nitrogen and eventually is purged by the passage of waste gases.

From the surge drum, waste gases are piped to one of two gas decay tanks and held up for approximately 60 days to reduce the activity. The decayed gas is then discharged, at a carefully controlled rate, to the suction side of the building exhaust fan. Interlocks automatically shut off the flow of waste gas if the fan stops.

The decayed gas and dilution air are thoroughly mixed during passage through the fan, after which the mixture is fed through the stack filtration system and discharged to the atmosphere. The flow diagram for this system is shown in Figure 5-4.

Waste gas from the radwaste liquid collection tank is pumped directly to the gas collection tank. Vapor from the top of the reactor tank is bled off, passing through the off-gas condenser to remove the water vapor, and then pumped to the gas collection tank. The condensate is fed to the liquid collection tank.

From the collection tank the gaseous waste is pumped to the waste gas surge drum and stored. When the pressure in the surge drum reaches 5 psig, gas is bled from the surge drum to reduce the pressure to 0 psig, and pumped to one of the two high-pressure gas decay tanks. Taps are provided on these tanks for radio-assay, and each batch will be analyzed periodically to determine whether or not controlled release can be permitted.

FLOW DIAGRAM
GASEOUS RADWASTE DISPOSAL SYSTEM

FIGURE 5-4

Gaseous Waste Component Description

1. Compressors

<u>Function</u>	<u>Power (hp)</u>	<u>Flow Rate (SCFM)</u>
Surge drum compressor	1.5	20
Hi-pressure compressor	6	20

2. Tanks

<u>Function</u>	<u>Capacity</u>
Waste gas surge drum	500 ft ³
Gas decay tanks (2)	50 ft ³
Gas collection tank	5 ft ³

3. Off-Gas Condenser

Design heat load - 2,000,000 Btu/hr
Design flow rate tube side (cooling water) - 250 gpm
Design flow rate shell side (condensing steam) - 3.5 gpm

Waste Gas Compressor: Both the surge drum compressor and the high pressure compressor have a capability of 20 scfm, the former operating at a head of 15 psia, and the latter at 250 psia.

Waste Gas Surge Drum: This tank is sized to hold 500 cubic feet of gas at 5 psig at room temperature.

Primary System Off-gas Condenser: When it is necessary to remove gases from the reactor vessel collection space or to deaerate the system, the accompanying water vapor will be condensed.

Gas Decay Tank: Each decay drum has a volume of ten cubic feet and is designed for a pressure of 250 psia.

5.4.5 Spent Fuel Pit Cooling and Purification System

Water in the spent fuel pit receives both heat and radioactivity from spent fuel stored in the pit. The function of the spent fuel pit cooling and purification system is to control the radioactivity content of and cool the spent fuel pit water.

System Description

Water is pumped from the pit by the spent fuel pit pump via a suction pipe which runs from the surface of the pit to within a few inches of the pit bottom. A vacuum breaker at the high point in the pipe prevents

the accidental emptying of the pit due to a pipe rupture or accidental valve opening. From the pump, the water is passed through the shell side of the spent fuel pit heat exchanger. The flow diagram for this system is shown on Figure 5-5.

Cooling is provided by the sea water cooling system.

Approximately five percent of the flow is diverted through the spent fuel pit ion exchanger and filter, thus maintaining a low activity level in the spent fuel pit water.

Component Description

1. Spent Fuel Pit Heat Exchanger

Quantity - 1

Design heat load - 8,000,000 Btu/hr

Design flow rate tube side (sea water) - 500 gpm

Design flow rate shell side (demineralized water) - 675 gpm

2. Spent Fuel Pit Pump

Quantity - 1

Design flow rate - 800 gpm

Power - 10 hp

3. Spent Fuel Pit

Diameter - 18 ft

Depth - 35 ft

Volume - 67,500 gal.

4. Spent Fuel Pit Filter

Quantity - 1

Design flow rate - 30 gpm

5. Spent Fuel Pit Demineralizer

Quantity - 1

Resin volume - 13 ft³

Design flow rate - 30 gpm

The spent fuel pit heat exchanger is designed to handle the heat load from a complete core twenty-four hours after shutdown, which is a heat load of approximately 8×10^6 Btu/hr. The required heat transfer surface is approximately 1650 square feet. The exchanger shell is carbon steel, with 70-30 cupro-nickel tubes.

**FLOW DIAGRAM
SPENT FUEL PIT COOLING & CLEAN-UP
SYSTEM**

FIGURE 5-5

The spent fuel pit pump is a 10-hp horizontal, centrifugal unit, designed for a flow rate of 800 gpm.

The spent fuel pit ion exchanger and filter are designed for approximately 30 gpm.

The pump and heat exchanger are located in the mechanical equipment area of the Reactor Building. The ion exchanger and filter are located in the shielded area below the spent fuel pit.

5.4.6 Decay Heat Removal System

The decay heat removal system is designed to maintain a main coolant water temperature of 125°F after the reactor has been shut down for approximately three hours.

System Description

Reactor coolant is pumped by the decay heat removal pump from the reactor tank top to the decay heat exchanger. From the heat exchanger, the cooled water is fed back to the bottom of the reactor tank through the purification system feed line.

A portion of the water leaving the pump, on the order of 100 gpm, is diverted and fed through the primary system purification system. This water returns to the suction side of the decay heat removal pump. The flow diagram for this system is shown on Figure 5-6.

Component Description

1. Decay Heat Exchanger

Quantity - 1

Design heat load - 20,000,000 Btu/hr

Design flow rate tube side (sea water) - 1200 gpm

Design flow rate shell side (reactor coolant) - 1700 gpm

2. Decay Heat Removal Pump

Quantity - 1

Design flow rate - 2000 gpm

Power - 30 hp

Three hours after shutdown, the decay heat production is approximately 5 Megawatts, or 17×10^6 Btu/hour. The decay heat exchanger is sized for a heat load of 20×10^6 Btu/hour. In order for the maximum temperature of the reactor coolant to be less than 125°F, the flow rate

FLOW DIAGRAM
DECAY HEAT REMOVAL SYSTEM

FIGURE 5-6

must be on the order of 1500 gpm. The decay heat pump is sized for 2000 gpm, with a required pumping power of 30 hp.

Both the decay heat pump and exchanger are located in the brine heater pit. The pump casing and exchanger shell are carbon steel, and the exchanger tubing is 70-30 cupro-nickel.

5.4.7 Component Cooling System

The component cooling system services those pieces of equipment which require fresh water cooling.

System Description

The system consists of a heat exchanger, pump, surge tank, cooling lines to the various components being cooled, piping, valves and instrumentation. Fresh water flows from the pump through the components being cooled, through the shell side of the component cooling heat exchanger, and back to the pump. The surge tank accommodates changes in water volume due to thermal expansion and contraction.

Component Description

1. Component Cooling Heat Exchanger

Quantity - 1

Design heat load - 400,000 Btu/hr

Design flow rate tube side (sea water) - 45 gpm

Design flow rate shell side (demineralized water) - 40 gpm

2. Component Cooling Pump

Quantity - 1

Design flow rate - 60 gpm

Power - 2 hp

3. Component Cooling Water Tank

Quantity - 1

Volume - 750 gal.

The heat exchanger, located in the equipment area, is designed to remove approximately 400,000 Btu/hour. The exchanger is a shell and tube type. Sea water circulates through the tubes, and component cooling water flows through the shell side. Flow rates are 45 gpm on the tube side and 40 gpm on the shell side. The shell is carbon steel and the tubes are of 70-30 cupro-nickel. The component cooling pump is a horizontal, centrifugal unit.

5.4.8 Emergency Poison Injection System

The function of the emergency poison injection system is to supply borated water to the core in the event that the control rods fail to maintain the reactor in a subcritical state.

System Description

The system consists of one 4000-gallon boron add tank located on the roof of the reactor building, one 8000-gallon secondary boron add tank located in the equipment area, a boron feed pump and associated piping and instrumentation. The flow diagram for this system is shown on Figure 5-7.

The boron in the 4000-gallon boron add tank, when uniformly distributed throughout the main coolant, would be sufficient to shut down the reactor if the rods fail to scram, with no xenon buildup, and with a fresh fuel loading at operating temperature.

The additional reactivity required to hold the cold, clean reactor shutdown with all rods out is supplied by the boron in the 8000-gallon secondary boron add tank.

Both tanks are filled with aqueous boric acid 15% solutions, heated to a temperature of approximately 150°F.

Upon receiving an appropriate command, which may be initiated either manually or automatically, valving will route boric acid solution from the boron add tank by gravity feed through a pipe running along the inner wall of the reactor tank and terminating at an exit nozzle located directly above the core. The boron thus introduced is carried by the coolant directly into the core, maintaining a negative reactivity. After shutdown, as the reactor temperature drops, it may be necessary to add additional boron to compensate for the reactivity change. This is accomplished by pumping solution from the secondary boron add tank to the core.

Component Description

1. Boron Add Tank

Quantity - 1

Volume - 4000 gal.

2. Secondary Boron Add Tank

Volume - 8000 gal.

FLOW DIAGRAM
EMERGENCY POISON INJECTION SYSTEM

FIGURE 5-7

FLOW DIAGRAM
LIQUID RADWASTE SYSTEM

FIGURE 5-3

3. Boron Feed Pump

Design flow rate - 500 gpm
Power - 25 hp

The tanks are horizontal right-cylinder cylinders. The boron feed pump is of the conventional horizontal centrifugal type, with a capacity of 500 gpm and a rated power of 25 hp.

5.4.9 Ventilation and Air Cooling System

The reactor building and pit ventilation and air cooling system serves several important functions. It provides ventilation of occupied spaces within the reactor building. It provides controlled flow of ventilating air from areas of low radioactive contamination toward areas of high or potentially high contamination. It removes heat from the reactor pit and the brine heater pit. It provides for positive containment of any radioactivity released from the primary system during abnormal or accident conditions. It provides a high-capacity vent from the brine heater pit to the stack to relieve pressure in the brine heater pit in the event of a main coolant system rupture. Thus, this system is an integral part of the reactor containment and is an important safety feature of the plant design. The system is shown schematically in Figure 5-8.

System Description

Normal ventilation flow goes from areas of low contamination to areas of high contamination to minimize the possibility of contaminating normally clean or uncontaminated areas with air that has passed through a contaminated area. Pressure differentials are also maintained such that leakage is also in the direction of increasing contamination, such as from outside the building inward and from the building into the reactor and brine heater pits.

Heat removal from the reactor pit is accomplished by once-through flow of ventilation air passing from the brine heater pit, through the reactor pit, to the exhaust blower, and up the stack. In the brine heater pit, the amount of heat to be removed is such that a recirculating air cooling system is provided.

The ventilation system consists of two 10,000-cfm exhaust blowers, each discharging to the ventilation stack, a filter system on each ventilation discharge, necessary ventilation ducting, and ventilation intakes provided with dampers. The exhaust blowers draw a total of approximately 15,000 cfm of air through the reactor building, approximately 4000 cfm of this from the administration building and control room, 1000 cfm from the brine heater pit cover and the remainder, or about 10,000 cfm, through ventilation intakes and through leaks in the reactor

building. The ventilation intakes have adjustable louvres so that a small differential pressure of approximately 1/4-inch of water can be maintained between the outside and inside of the building. A flow of approximately 1000 cfm from the radwaste vault below the equipment space is ducted directly to the ventilation discharge without passing into the reactor building since the radwaste vault is a potentially contaminated area.

Of the total air drawn into the reactor building, approximately 6000 cfm is exhausted directly to the stack through one of the two exhaust blowers. Approximately 8000 cfm is drawn by the other blower through an intake into the brine heater pit, from the brine heater pit to the reactor pit through a connecting duct at the bottom of the pits, from the top of the reactor pit, and through the filter system to the stack.

A separate ventilation intake for the reactor pit and suitable valving is provided so that air can be drawn directly into the reactor pit from the reactor building. This permits the reactor pit to be ventilated separately from the brine heater pit when the brine heater pit is open for maintenance and controlled ventilation must be maintained in the reactor building, such as during refueling.

Filter systems consisting of roughing filters, absolute filters, and charcoal filters are provided on each ventilation duct upstream of the exhaust blowers. The ventilation flow from the reactor pit, the brine heater pit, and the radwaste vault normally passes through the filters. The flow from the reactor building normally bypasses the filter system but is diverted through it on a signal from the activity monitors in the stack and in the flow stream upstream of the filters. The two ventilation ducts are cross-connected and excess blower capacity is provided so that one blower can serve both ventilation functions, at reduced capacity, in an emergency or so that either filter system can be used for either flow stream if necessary.

The brine heater pit cooling system consists of a closed air circulation loop through which brine heater pit air is circulated at a rate of approximately 46,000 cfm. Air is drawn from the top of the pit and returned to the bottom after passing through a finned-tube heat exchanger cooled with sea water. The total heat load due to losses from the brine heaters, brine piping, and main coolant piping is approximately 570,000 Btu/hour. The temperature of the brine heater pit air is maintained at approximately 110°F with this system. The temperature of the air in the reactor pit is also maintained at approximately 110°F by the ventilation flow of 8000 cfm of air through the pit.

Discharges from the gaseous radioactive waste system are released directly to the stack where they are diluted by the total ventilation system flow of approximately 20,000 cfm (including outside air ducted

through dampers to the exhaust blower inlets to permit the ventilation system to be properly controlled).

A large, normally closed, duct connects the brine heater pit directly to the stack to vent the steam which would be released in the event of a main coolant system rupture. The initial steam release rate immediately upon rupture could be as much as 300 lbs/sec. (approximately 8000 cfs of steam at atmospheric pressure). The valve normally closing the relief duct is opened on over-pressure in the brine heater pit and closes again to retain controlled ventilation when the steam release rate has reduced sufficiently to permit the pit ventilation system to operate normally.

Component Description

1. Brine Heater Pit Ventilation Blower

Quantity - 1
Design flow rate - 46,000 cfm
Power - 50 hp

2. Brine Heater Pit Heat Exchanger

Quantity - 1
Design heat load - 570,000 Btu/hr
Design flow rate (sea water) - 120 gpm
Design flow rate (air) - 46,000 cfm

3. Ventilation Exhaust Blowers

Quantity - 2
Design flow rate - 10,000 cfm
Power - 20 hp

4. Roughing Filters, Absolute Filters and Charcoal Filters

Quantity - 2 each
Flow area - 50 ft²

5.4.10 Sea Water Cooling System

The function of the sea water cooling system is to supply the cooling water to the various auxiliary system heat exchangers.

System Description

The system supplies cooling water to the following heat exchangers: component cooling, brine heater pit, off-gas condenser, spent fuel pit,

decay heat removal, and the purification system non-regenerative heat exchanger. In addition, when an outage occurs in the 200-kv electrical power supply, the system (which is ordinarily powered from the 33-kv power supply) supplies cooling water to the brine side of the third-stage brine heater. This brine heater is then used, in conjunction with a main coolant pump, to cool down the reactor and the first- and second-stage brine heaters.

Component Description

Sea Water Pumps

Quantity - 2

Design flow rate - 2000 gpm

Power - 25 hp

5.5 BUILDINGS AND STRUCTURES

5.5.1 General

A single integrated structure houses the reactor, spent fuel storage pit, radwaste storage areas, equipment area, and administrative and control areas. The brine heaters, main coolant pumps and related equipment are located in a pit with a removable cover immediately adjacent to this structure.

5.5.2 Design Criteria

Codes

The structural design of the plant would be in accordance with the current Uniform Building Code and its appropriate reference codes.

Seismic Loads

All structures would be designed for seismic loading in accordance with the latest issue of the Uniform Building Code or other competent authority. The proposed design uses a lateral and vertical seismic loading of 0.33g, as explained in Section 3, SITE DESCRIPTION.

Wind Loadings

All structures would be designed for wind loadings in accordance with the latest issue of the Uniform Building Code or other competent authority. The proposed design is based upon recommendations of the ASCE Task Committee on Wind Forces (Cf. Transactions, ASCE, Vol. 126, Part II, 1961, pp. 1124-1198). The present analysis uses a 100-year design wind force of 80 mph.

Reactor Building Live Loads

Ground floor except laydown areas	500 psf
Equipment areas	250 psf
Roof areas	20 psf
Stairs and platforms	100 psf
All floors used as laydown areas	600 psf

Foundations

Depths of foundations and allowable soil bearing capacity would be determined by field and laboratory test of soil samples. Foundation bearing capacity, based upon consideration of typical site geology, is estimated to be 10 tons/ft².

5.5.3 Reactor and Brine Heater Pits

The reactor pit has a 15-foot inside diameter and the brine heater pit has a 42-foot inside diameter. Both pits have two-foot thick concrete liners. Where this lining coincides with concrete of the caisson or shaft liner in which the two pits are enclosed, the caisson concrete liner is used as the pit liner. The caisson liner is reinforced concrete 6 feet 9 inches thick, and is lined with a 1/4-inch thick carbon steel liner. The bottom plug is eight-foot thick reinforced concrete with a 1/4-inch carbon steel liner which is joined to the wall liner to completely waterseal the pits. The bottom plug acts as a common foundation for both the reactor and brine heater pits. The top of this plug is 100 feet below sea level. The two pits are inter-connected by piping holes along the line of tangency at points 4 feet and 40 feet above the pit floor.

The two-foot thick concrete liner of the reactor pit extends upward to elevation +50.

The brine heater pit is covered with four-foot thick reinforced concrete shielding beams. The beams are covered by a movable shed which has a steel frame with fluted metal panel covering. This shed is rail-mounted so that it can be easily removed for access to the shielding blocks and pit below. The shed is designed for limited leakage so that the controlled ventilation system can ensure that leakage is inward.

A 125-ton truck mounted crane (available from rental agencies in southern California) is required to remove the concrete shielding blocks and place them in a laydown area provided. Maintenance personnel can enter the brine heater pit through a manhole and tunnel from the reactor room. A removable metal grating floor is located at pump elevation within the brine heater pit.

5.5.4 Spent Fuel Storage Pit

A 15-foot diameter concrete cylinder, 38 feet deep, is provided for storing spent fuel and loading the spent fuel shipping cask. The water depth in this stainless steel-lined cylinder will be maintained at 35 feet. The two purification system ion exchangers and the spent fuel pit ion exchanger are located in the enclosed area underneath this tank.

5.5.5 Reactor Building

The above-ground portions of the reactor and the spent fuel pit will be housed in a high-bay area 30 feet wide, 60 feet long, and 78 feet high to the top of the walls. The roof slopes at a 30-degree angle towards either side of this area. One end of this room is a combination laydown area and loading dock. The building is covered with fluted metal panels which are caulked to minimize building leakage. A 30-ton bridge crane with 5-ton auxiliary hoist is mounted on a rail 72 feet above the floor of the high bay area. The main lifting hook is required for handling the spent fuel cask and the reactor pit cap, while the auxiliary hook is used for lifting fuel elements and other lighter loads, such as the manhole cover to the brine heater pit access.

A 30 by 60-foot low-bay area is provided adjacent to the high-bay area. This low-bay area contains pumps, compressors, blowers and storage tanks for non-critical liquids. This area is covered with fluted metal panels.

A vault for storage of radioactive wastes is provided in a basement below the equipment area. This vault is concrete shielded. Access to the vault is provided through manholes.

The administrative offices, control room, staff comfort areas, monitor and dressing rooms, laboratories and other administrative and control areas for the entire facility are located in a two-story portion of the structure. This area will be fully insulated and air conditioned for comfort. Exterior cover will be metal siding. Electrical switchgear for the reactor facility will be located in this area immediately below the control room.

5.5.6 Ventilating Stack

A concrete stack, 200 feet high and 16 feet in diameter at the base, is provided adjacent to the reactor facility for discharge of ventilating air and gaseous radioactive wastes.

5.6 AREAS REQUIRING SPECIAL CONSIDERATION

5.6.1 General

As indicated in Section 1, the plant design is based, to the maximum extent consistent with low cost, on proven concepts and current technology. No credit is assumed for technological break-throughs or future developments which could further simplify the design and reduce plant cost. Where uncertainties have arisen due to the unique features of the plant, conservative or middle-of-the road assumptions were made rather than optimistic ones. Consequently, it is felt that a reactor plant of this design is technically feasible and could be built with a limited amount of equipment development and proof testing.

In this section, special attention is devoted to the areas of the conceptual plant design which deviate the most from other reactor plant designs or which are based on the least past experience. While uncertainty remains in these areas and some development and testing work should be performed for the first plant of this type, there is no reason to believe that the uncertainties cannot be satisfactorily resolved.

5.6.2 Reactor System Materials and Chemistry Control

Selection of materials of construction for any facility is a trade-off between performance and cost. Factors which must be considered include material strength and corrosion rate under conditions of use, cost of fabrication and of future replacement, and basic material cost. For reactor plants, consideration also must be given to problems of radiation which may affect material properties and accessibility to the system for maintenance and repair. For most reactor plant designs, consideration of these factors has led to the selection of stainless steel as the basic primary system material. However, for the deep-pool reactor system, several considerations led to the rejection of stainless steel as the basic reactor system material:

1. The presence of sea water surrounding the plant and the possibility of sea water leakage into the reactor system through the brine heaters and other sea-water-cooled heat exchangers raise concern about chloride stress corrosion of stainless steel.
2. The system operates at a low temperature and pressure (270°F and 112 psig, maximum) so that special materials or fabrication techniques should not be required.
3. With low cost as a design objective, the large volume of the reactor system, particularly of the reactor tank, makes materials costs disproportionately high unless a low cost material were used.

Other materials which have been used for the construction of reactor systems are aluminum and carbon steel. In the initial COO study, aluminum was first proposed as the basic primary system and fuel cladding material to take advantage of its low cost, corrosion resistance, and its particular suitability for low temperature and low pressure operation. However, aluminum was not recommended principally because of the possibility of accelerated corrosion by copper ions from the brine heater tubes. If aluminum is used, brine heater tubes would have to be of some other material, such as titanium, having a low corrosion rate in brine but not containing copper. Titanium tubing is still more expensive and has had much more limited use than copper alloys for high temperature sea water heat exchangers. The cost is expected to decrease as its use becomes more widespread and it should be considered as a brine heater tube material in future plants. Because of the very low corrosion rate of titanium, the tubes probably would not have to be replaced during the lifetime of the plant, a significant advantage in a reactor plant where radiation levels will complicate maintenance.

The choice of carbon steel for the reactor plant was based on minimizing overall plant costs consistent with satisfactory performance based on current technology. The primary area of concern in a carbon steel plant is the corrosion rate. The large surface area of this system makes the corrosion rate particularly significant. For a reactor system, even very low levels of corrosion are of concern since the activated corrosion products ("crud") formed by neutron activation in the core deposit on system surfaces, thereby increasing problems of accessibility and maintenance. Crud also constitutes a major radioactive contaminant of the liquid wastes which must be discharged from the plant. The coolant purification system provided to maintain high reactor coolant purity and low crud levels increases in size in proportion to the main coolant system volume, surface area and corrosion rate. However, carbon steel reactor systems have been operated successfully and corrosion rates held to acceptable levels. Examples include the Hanford New Production Reactor (NPR), the Canadian Nuclear Power Demonstration Reactor (NPD), the Heavy Water Components Test Reactor (HWCTR), and the steam or feedwater piping of numerous boiling water reactors. The Hanford out-of-pile and in-pile experience (10, 11, and 12) leading to the use of carbon steel piping in the NPR can be summarized as follows: A uniform corrosion rate of less than 0.2 mil/yr can be maintained in deoxygenated, demineralized light water at temperatures up to 290°C with a pH of 10 and LiOH resin. Comparable results are obtained with ammonium hydroxide resin pH control. There are some data which suggest that at lower temperatures a lower pH may be satisfactory.

The Canadian NPD experience with carbon steel piping was recently reported in reference ¹³. Normal crud concentration in the heavy water was less than 0.01×10^6 ppm with peak concentrations a factor of 10

higher. The pH was maintained at 10-11 using LiOD and the dissolved oxygen was kept less than 0.01×10^6 ppm. The system contains approximately 3500 ft² of carbon steel and was operated at 530°F.

In the HWCTR, primary piping and steam generator and heat exchanger tubes are carbon steel. Operating experience ⁽¹⁴⁾ may be summarized by noting that the release rate of iron to the heavy water coolant was 16 mg/dm²/mo. (approximately 0.1 mil/yr) for exposure up to 4400 hours. Coolant conditions were 200 to 250°C, a pD of 10 - 11 with LiOD, and 14 ppb dissolved oxygen. In these plants the carbon steel surfaces were conditioned prior to reactor operation by circulating hot deoxygenated water to form an adherent Fe₃O₄ coating on the surfaces.

Based on the above it appears that the corrosion rate of a carbon steel reactor system can be controlled within acceptable limits by proper reactor coolant chemistry control. By maintaining the level of oxygen in the coolant to very low levels (approximately 0.1 ppm or less) and pH in the range of 10 to 12, acceptably low corrosion rates should be achievable. Consequently, the reactor system is designed so that the reactor coolant can be maintained at a high pH and at a low oxygen concentration. To prevent air from entering into the main coolant system, the reactor tank is provided with a sealed lid and the control rod drive mechanisms mounted on the lid are sealed units.

Excluding air from the reactor system may not be sufficient, however, to ensure low oxygen level in the coolant. Because of the oxygen and hydrogen formed by radiolytic decomposition of the coolant water, oxygen might build up to significant levels. Therefore, it was considered desirable at this point in the design to provide a coolant deaeration system to remove oxygen and other dissolved gases from the coolant. This system sprays reactor coolant into the space below the reactor tank lid where a portion of it flashes to steam. The vapor from the space is drawn off through an off-gas condenser where the water vapor condenses and is sent to the liquid radwaste disposal system while the non-condensable gases are sent to the gaseous radwaste system. This system also provides a means for removing and controlling fission product gases which might be released to the main coolant.

Hydrazine has been added to reactor coolant systems to scavenge oxygen in some cases, particularly to reduce the dissolved oxygen content after initial fill and before reactor startup. It also could be used periodically during plant operation. However, if ammonium hydroxide resin is used for pH control, as discussed below, it serves the same function as the hydrazine. Moreover, since hydrazine forms ammonia after reacting with oxygen, if the corrosion of copper alloys precludes the use of ammonium hydroxide, as discussed below, hydrazine would also be pre-

cluded. Other oxygen scavenging additives such as sodium sulfite have been widely used for boiler water treatment but are not recommended for reactor systems because of the activation products which may be produced as the coolant flows through the core.

The pH of the reactor coolant is controlled by use of an appropriate ion exchange resin in the coolant purification system. Potassium, ammonium, and lithium hydroxide resins have been used. Of these, lithium hydroxide will result in the highest pH and is frequently used for this purpose. Natural lithium (predominantly Li-6) produces tritium in an (n, α) reaction but Li-7 resin is available which avoids this difficulty. Ammonium hydroxide resin has the principal advantage of scavenging oxygen from the system in addition to maintaining high pH. In a radiation flux, the ammonium ion apparently forms hydrazine which is frequently used as an oxygen scavenger. Ammonium resins which are capable of maintaining a pH above 10 are available. For this reason, ammonium hydroxide resin has been assumed as the principal main coolant chemistry control agent in this plant. There is, however, some uncertainty about the use of ammonium ions with the copper-bearing brine heater tube materials. Ammonia forms a complex ion with copper which could accelerate the corrosion of copper-bearing materials. There is also some evidence that ammonia causes stress corrosion cracking of copper materials. It is believed that these reactions occur only in much higher concentrations of ammonium ion than would be present in the main coolant system and that they occur only with much less corrosion resistant copper-bearing materials than the 70-30 cupro-nickel used for the brine heater tubes. If corrosion of the reactor coolant side of the brine heater tubes appear to present any sort of difficulty, lithium hydroxide resin could be used for pH control and the oxygen maintained at low levels by means of the deaeration system.

The choice of 70-30 cupro-nickel for the brine heater tube material was based on obtaining the maximum corrosion resistance in reactor coolant with a material proven to be corrosion resistant in brine. Brine heater tubes normally are made of a high copper alloy such as aluminum brass or cupro-nickel to provide necessary corrosion resistance in brine at reasonable cost. As mentioned previously, titanium also has a low corrosion rate in brine but limited operating experience and higher cost ruled it out for this study. Although the high copper alloys are known to have low corrosion rates, there is little quantitative information on exactly how low these rates would be in a reactor coolant system. Therefore, 70-30 cupro-nickel was selected because of this uncertainty, because of the large surface area of the brine heater tube material which must resist corrosion, and because its general corrosion resistance is superior to that of the other high-copper alloys.

The reactor system as designed provides several options to the operators for maintaining proper coolant chemistry and low corrosion rates of the main coolant system materials. Without further testing it is difficult to predict at this time the extent to which the various methods would be used.

Based on results of corrosion rate tests in conditions similar to those in the main coolant system, the coolant purification system has been designed assuming a corrosion rate of 0.5 mil/yr for the carbon steel surfaces and a rate of 0.1 mil/yr for the 70-30 cupro-nickel brine heater tubes. These rates are felt to be conservative but must be assumed to be rough approximations only, since corrosion data is not available for the exact conditions of oxygen concentration, temperature, pH, etc., which will exist in this plant. Experience at Hanford, previously cited, indicates that corrosion rates of carbon steel as low as 0.2 mil/yr can be maintained consistently in a closed carbon steel reactor system under somewhat similar conditions. It is generally considered that 70-30 cupro-nickel has a corrosion rate much less than that of carbon steel but for most commercial applications a material is considered to be "corrosion resistant" if the rate is less than 2 mils/yr. Therefore, little information is available on exactly how low corrosion rates of cupro-nickel are. Even at the low rate assumed, the corrosion of the cupro-nickel is the controlling factor in determining the size of the coolant purification system because the brine heater tubing constitutes a large proportion of the main coolant system surface area (approximately 300,000 ft² of cupro-nickel vs. 20,000 ft² of carbon steel). Thus, any increase in the corrosion rate of cupro-nickel above that assumed would increase the size of the coolant purification system and cost of ion exchange resins. The converse is also true.

Because of the use of carbon steel as the principal material of construction, because of the large volume and surface of this system, and because of the limited information available on the corrosion rate of the reactor system materials under the assumed conditions, the area of materials corrosion and coolant chemistry control remains one of uncertainty. However, from the information available it would appear that a rather limited test program can resolve these uncertainties. Further investigation is required as follows:

1. The general corrosion rates of carbon steel and cupro-nickel should be confirmed for the conditions of this reactor. It does not appear that the feasibility or basic design of the system would be materially affected; however, the size of the coolant purification system could be altered.
2. The effects of ammonium hydroxide on the corrosion rate of cupro-nickel should be determined. As noted above, if adverse

corrosion effects are found to occur, lithium hydroxide or some other pH control agent could be used or non-copper-bearing alloys substituted for brine heater tube materials.

3. The possibility of galvanic corrosion of the carbon steel in the reactor system should be investigated. The carbon steel would be expected to be anodic to the cupro-nickel brine heater tubes in which case its corrosion rate could be somewhat accelerated. Because of the large ratio of cupro-nickel surface area to carbon steel surface area (15 to 1), it is conceivable that considerable acceleration of corrosion might take place. However, in an in-pile loop test at Hanford in which a cupro-nickel tube bundle was placed in a carbon steel loop under reactor operating conditions, no accelerated corrosion was apparent ⁽¹⁵⁾. A test directed specifically at this question and with proper area ratios and coolant chemistry conditions should be undertaken.
4. The need for, and performance requirements and design details of, the coolant deaeration system should be determined. For example, the recombination rate of hydrogen and oxygen produced by radiolytic decomposition should be determined for the appropriate conditions and the possible need for a catalytic recombiner should be investigated.

5.6.3 Control Rod Actuation

Because of the depth of the reactor tank, the method of actuating the control rods requires special consideration. Several alternate concepts for control rod actuation in water reactor systems are presently in use but none requires the control rod drive mechanism to be located 135 ft or so from the core itself as in this concept. Thus, the control rod drive arrangement assumed for the present design must be considered somewhat of a departure from present practice. Reasons for selecting the top-mounted, magnetic jack drives were given in paragraph 5.2.2. Several alternatives, all adaptations of currently used techniques, were considered, again with the objective of making maximum use of existing technology. The principal alternatives and the basic considerations for and against each are outlined below:

Location of Drive

Bottom-mounted Drives: Bottom-entering rods with the control rod drive mechanisms located beneath the reactor tank were considered. This concept, similar to that used on present-day boiling water reactors, has the advantage of locating the drive near the core and of allowing access to the top of the core for refueling. However, the requirement for an accessible area beneath the reactor would require that a sub-

stantially deeper hole be excavated, and special precautions would be required to ensure that ground water could not leak into the sub-pile room or that the drives could scam the reactor in the event that the room should become flooded.

Side-mounted Drives: Rod drive mechanisms located at the side of the reactor tank at core height could be used to drive the rods vertically out of the core through a gear train arrangement. This concept has the advantage of placing the drive mechanism near the rods to shorten the gear train but also requires considerable additional excavation to provide an accessible area in which the mechanisms could operate.

Type of Drive

Cable Drives: Cable drives are probably the simplest form of control rod actuation in current use, particularly for pool-type reactors. This concept was assumed in the original COO-278 report. While the concept has the advantage of simplicity and can certainly be adapted to the reference design, it was rejected in favor of other methods for several reasons. It was felt that in a power reactor it is desirable to have the capability of positive drive-down of the control rods rather than depending entirely on gravity insertion. Cable drives also present greater difficulties in latching and unlatching the rods for refueling in view of the depth involved.

Rotary Motion: To provide a means for positive insertion and still have the drive mechanisms mounted at the top of the reactor, a rotational drive through a gear train, perhaps to a rack and pinion drive at the top of the core, was considered. The concept appears feasible and the individual components required are standard items, but the uncertainties involved in operation of gears in water, the problems due to accumulation of crud, misalignment tolerance requirements, and the difficulties with removing the drive components for refueling appeared to require more development testing than other concepts considered.

Hydraulic Ball Control Concept (Hy-Ball): The hydraulic ball concept for reactor control, described in report SROO-207-30⁽¹⁶⁾ might be considered for this reactor because of the unique requirements of the concept. However, the Hy-Ball concept was not evaluated for this application because it has not been used in a reactor and therefore was not considered existing technology.

Hydraulic Drives: Hydraulic control rod drive mechanisms have been used extensively and would be feasible for this application. Hydraulic drives could be considered to provide linear motion from the top of the reactor tank or within the reactor tank just above the core level. The last concept presents the same difficulties of accessibility for mainte-

nance and removal for refueling indicated for the gear drive concept mentioned above. The principal difficulty with linear motion hydraulic drives appears to be that of position indication and drift.

The extension pipes extending from the rods to the top of the tank in the reference design are a departure from current practice but appear to present no insurmountable difficulties. During all aspects of normal operation the pipe is in tension. Even without lateral support of the extension pipes between the top of the core and the top of the reactor tank, a three to four-inch diameter pipe can transmit substantial downward force without buckling. Because of the weight of the extension pipe, it is probably not necessary to apply an additional downward force for positive control rod insertion. The extension pipe also is capable of transmitting a rotary motion which would facilitate rod latching and unlatching. During refueling operations, the control rod extension pipes are unlatched and temporarily stored out of the way around the side of the reactor tank.

To provide a degree of lateral support of the control rod extension pipes, retractable arms are located just above the reactor inlet pipe on the reactor tank wall. These two sets of arms, located at 90° to each other, will fold against the reactor tank wall during refueling operations. In the installed position they form a grid to provide lateral support of the extension pipes in either direction and also substantially increase the compressive load which could be placed on the extension pipe if necessary. This support grid serves as a partial flow barrier to deflect the main coolant flow downward and reduce somewhat the convective heat transfer to the surface of the reactor tank.

A detailed evaluation of the control rod drive alternatives was not made during the study but it appears that several approaches are feasible within existing technology and should require only a minimum of equipment development and component testing prior to construction of a reactor prototype. The approach selected for the conceptual design was selected primarily as a basis for developing a cost estimate and should not necessarily be considered the best of the approaches considered and listed above.

5.6.4 Refueling

Refueling is an area requiring special consideration only because of the depth of the reactor tank. Fuel handling through substantial depths of water is common practice. The difficulties associated with this operation are magnified in going to the greater depth (about 134 feet to the top of the core for this reactor). New techniques or special adaptations of existing techniques may be required. The performance requirements for the reference concept considered in the study are described below.

A means of positively locating a grappling tool over a given fuel element would be required. Underwater television cameras probably would be necessary. In addition, the grappling device must have sufficient rigidity so that it can be positioned accurately at the core by moving the end at the top of the reactor tank. To avoid the necessity of withdrawing the entire positioning device from the tank to transfer a fuel element, a refueling dolly or elevator would be provided at the side of the tank to lift the withdrawn fuel element to near the top of the reactor tank. The positioning device need have capability only for sufficient vertical motion to remove the fuel element from the core and transmit it laterally to the refueling dolly. Then, as the dolly lifts the fuel element to the top of the tank, the positioning device can be positioned over another fuel element. Near the top of the tank a second hoist, such as the auxiliary hoist on the overhead crane, can take the fuel element out of the refueling dolly and transmit it laterally to the spent fuel pit adjacent to the top of the reactor tank.

This refueling concept permits complete underwater handling of fuel and thus reduces the probability and severity of fuel handling accident. By lifting fuel in the reactor tank at the same time that the next fuel element is being located and grappled, a substantial savings in refueling time can be realized. The deep-pool reactor concept also offers some simplicities which should reduce refueling time. For example, the low operating temperature reduces the time required for cooldown before refueling. The unpressurized system requires less time to open up before refueling and to close up after refueling and it would not require high pressure hydrostatic pressure testing before plant startup.

Since underwater fuel handling operations have not been performed in such deep water, equipment development and underwaterproof testing in a full-length mock-up should be performed. However, it appears that refueling operations should be inherently no more difficult than for other types of reactors.

5.6.5 Containment Criteria

Special consideration should be given to the requirements for containment to take maximum advantage of the inherent safety characteristics of the deep-pool reactor concept. Because of its low operating temperature and pressure, and consequently the low stored energy in the coolant, and because the reactor is located below a large reservoir of water and thus cannot sustain a loss-of-coolant accident, it appears that the plant could be located in areas of relatively high population density without the need for conventional pressure-tight reactor containment. This should result in substantial cost savings while at the same time reducing concern about the consequences of severe nuclear accidents. Because of the importance of reactor safety to plant design, careful consideration of reactor containment and engineered safeguards is required.

The hypothetical accident resulting in the most serious consequences for most water-cooled nuclear power plants occurs when the main coolant system is ruptured, the coolant power expelled from the system, the core uncovered, and some portion of the fuel melted, thereby releasing a large fraction of the fission products contained in the fuel. Because of this possibility, even though it is very remote, it has been necessary to provide pressure-tight and essentially leak-tight containment structures around most nuclear plants. The pressure for which these containment structures must be designed is dependent upon the stored energy in the coolant which, in a water-cooled reactor, will cause a large portion of the coolant to flash to steam. The leak-tightness requirement depends upon the quantity of fission products which could be released from the reactor in an accident, as determined by an accident analysis, the location of the reactor plant with respect to areas of population, and other factors such as meteorology. Most nuclear plants have containment vessels with leakage rates on the order of 0.1 percent of the contained air per day. More recently, containment concepts have been developed for populated areas which employ two containment barriers and with leakage rates considered to be essentially zero. For the deep-pool reactor, the type of loss-of-coolant accident and resulting melting of fuel described above is considered to be incredible. The entire main coolant system is located below grade in two concrete-lined pits. The free volume of these pits is such (or can be made such) that if the main coolant system is ruptured, the resulting level of main coolant water in the pits would be well above the level of the core. Furthermore, the reactor is located well below the ground water level so that even if a major catastrophe should cause the pit walls to rupture, the core would remain covered with water. Removal of decay heat from the core could be accomplished by boiling of the water surrounding the core if no other means were available. Because of the large volume of water available, there would be ample time for corrective action to be taken before any possibility of uncovering the core would exist. Because of the low stored energy in the main coolant system, only about 30,000 lb out of the total of nearly 2.5×10^6 lb of coolant would flash to steam after the rupture. This amount of steam could be vented to the stack without over-pressurizing the pits and the amount of activity carried out by the steam or released subsequently from the fuel is quite low.

Since melting of the fuel is not considered possible, only a very small fraction of the contained fission products would be released in the event of any accident. The principal sources of radioactivity which could be released are the activation products which could be present in the reactor coolant and fission products which could escape from the fuel through defects in fuel cladding. Therefore, a means should be provided to collect and process for disposal the air in any space to which this radioactivity could be released. Controlled ventilation, filtration, storage for decay, and a stack for elevated discharge are considered to be the principal containment requirements for this reactor concept.

A number of controlled ventilation features have been provided. The top of the reactor tank is sealed with a lid, primarily to prevent air from entering the main coolant system and increasing corrosion but also to provide a means for collecting any radioactive gases released from the main coolant. The space below the lid is evacuated to the gaseous radwaste disposal system. The reactor pit and the brine heater pit are isolated from the surrounding buildings and provided with independent controlled ventilation. This ventilation serves both to remove heat from the pits and provide sufficient flow rate to ensure that all leakage is inward. The exhaust from the pit ventilation is directed through a filter system and up the stack. A separate controlled ventilation system is provided for the reactor building covering the two pits and provides essentially a second containment barrier. This ventilation flow also is sufficient to ensure that all leakage is inward and also is discharged up the stack.

The controlled ventilation containment concept (sometimes referred to as "confinement") has been used for other reactors where there is little stored energy in the coolant, such as for pool-type research reactors. The Oak Ridge Research Reactor is an example⁽¹⁷⁾. However, the concept has not yet been used for reactors with high power levels or reactors located in areas of relatively high population density such as the plant described in this report. More detailed studies of the type indicated in Section 8 will be necessary to demonstrate the safety and reliability of this containment concept for use with the deep-pool reactor. It also would be advisable to informally review the assumptions and conclusions regarding safety in this report with the AEC regulatory authorities before committing extensive design and development. However, as the results of these preliminary studies indicate, even very pessimistic assumptions lead to almost inconsequential results. Indeed, it is believed that the single most attractive feature of the deep-pool reactor concept for the application described in this report is its inherent safety.

Section 6

WATER PLANT

6.1 GENERAL

The single-effect, multistage flash evaporation process, utilizing recycle of acid-pretreated brine, is presently considered to be the best proven process for large water desalting plants and is the basis of the water plant conceptual design in this report. Since, in this study, the main emphasis was on the reactor plant, the water plant design was based primarily on results of previous studies. The design was developed during the study only to the extent necessary to adapt the process to the unique features of the reactor plant and to obtain reasonable cost estimates.

In this study, a nominal 50-mgd plant was used as the reference case. From this reference case, alternates were developed for a 10-mgd and a 200-mgd plant. The reference case is described in this section; the other plants are noted only where their features would differ from those of the reference plant. The design criteria for the reference plant are shown in Table 6-1.

This section briefly describes the multistage flash evaporation process and principal plant equipment and outlines the method of plant optimization used to establish the basic water plant design parameters.

6.2 PROCESS DESCRIPTION

The multistage flash evaporation process is illustrated schematically in Figure 6-1. The flow rates, temperatures, and number of stages indicated in this diagram apply specifically to the first section of the reference 50-mgd plant but the process is typical for all sections. In this process, the incoming sea water is slugged periodically with chlorine to prevent the growth of algae, barnacles, and other marine life. It is then split into two streams, one to provide make-up water to the basic process and the other to provide cooling water for the heat-rejection section of the plant and for auxiliary services, such as the air ejector condensers. The pH of the make-up water is adjusted with sulfuric acid for the decomposition of carbonates, after which degasification is accomplished in an open tank. The pretreated water then enters the tubes of the condensers in the heat-rejection section of the plant, where it is heated to the same temperature as the concentrated brine leaving the plant via the last stage.

TABLE 6-1

DESIGN CRITERIA FOR MULTISTAGE FLASH EVAPORATOR
50 MGD PLANT

Capacity	50 mgd
Heat Supply at Brine Heaters (From Reactor Coolant)	400 Mwt
Maximum Temperature of Reactor Primary Coolant	270°F
Minimum Temperature of Reactor Primary Coolant	200°F
First Stage Brine Heater Maximum Temperature	250°F
Second Stage Brine Heater Maximum Temperature	222°F
Third Stage Brine Heater Maximum Temperature	201°F
Brine Temperature Rise across Each Heater	11°F
Sea Water Entering Temperature	60°F
Sea Water Salinity	35,000 ppm
First Stage Concentrating Ratio	1.7
Second and Third Stage Concentrating Ratios	2.0
Maximum Product Water Temperature	85°F
Maximum Product Water Salinity	10 ppm
Product Water Delivery: to site boundary at a pressure head of	800 ft
Difference in Elevation between Hot and Cold End of Water Plant	23 ft
Maximum Permissible Brine Velocity in Tubes	7 fps
Tube Material for Evaporators	70-30 cupro-nickel
Tube Material for Brine Heaters	70-30 cupro-nickel
Tubesheet Material	Naval rolled brass
Evaporator Design: similar to Point Loma I but without experimental features	
Vessel Construction Material: Steel with 1/8-in. corrosion allowance.	

Finally, it flows into the deaeration chamber of the last stage, where it is stripped of non-condensables by a counterflow of steam, originating from the flashing brine at the bottom of the stage and then is mixed with the brine.

Prior to this mixing, a flow of brine equal to the difference between the make-up water flow and the product water flow is rejected by means of the blowdown pump. This is the sole means of salt removal from the plant. The remaining brine, along with the make-up water, constitutes the recycle brine stream. This stream flows through the condenser tubes of the heat recovery section of the plant, countercurrent to the flashing brine, and serves two basic functions:

1. It condenses the water being produced.
2. It recovers heat and reduces the required reactor heat rate.

After having been pumped through the condenser tubes and the brine heater by means of the recycle pump, the brine enters the first, or hottest, stage of the evaporator to begin a series of flashings. The flowing stream takes its pressure drops through underflow weirs in the barrier walls, with orifices properly sized for each stage. From the final evaporator stage it returns to the vacuum sump for recycling.

The product water is collected in troughs located beneath the condenser bundles which convey it from higher to lower pressure stages via internal loop seals. The product water releases its sensible heat to the in-tube brine by flashing and recondensing on the tubes as it passes from stage to stage. The product water is separately collected from the heat recovery and heat rejection sections of the plant, resulting in substantial savings in heat rejection tube area and in pumping power requirements compared to total plant collection systems. The distilled product is metered, checked for purity by conductivity meters and radiation monitors, and is finally pumped to the plant boundaries for delivery to storage facilities.

6.3 GENERAL PLANT DESCRIPTION

6.3.1 Evaporator System

The evaporators are a series of rectangular steel vessels, each containing interior bulkheads which divide the vessel into a number of separate stages. Each stage provides area for flashing brine, vapor space for entrainment separation and for condensation of the product on the recycle brine tubes, and troughs to collect and convey the product water. Entrainment separation is accomplished by suitable means, such as demisters, within each stage. Flash vaporization is

accomplished by providing a residence time of about 5.5 seconds within each stage. The brine is about 12 inches deep. With the widths shown on the Site Plan, Drawing EA1, the stages average about 12 feet long in the direction of brine flow. To provide sufficient volume for the condensing tube bundles, the stages are approximately eight feet high. Baffled offtakes for removal of non-condensables and splash plates for deflecting the condensate to prevent flooding of the tube surfaces are located within the tube bundles.

Transfer of the hot flashing brine from stage to stage is governed by interstage orifices in the bulkheads, followed by suitable baffles. For the flashing product, interstage loop seals provide the necessary flow continuity and pressure drops. At the ends of vessels, flow continuity for all streams is provided by means of water boxes, valves, piping and headers. Access to the heads of tube bundles for servicing is provided by placing each vessel at a lower level than the one preceding it. This arrangement also provides a hydrostatic head between vessels to insure a positive flow of brine in the lowest stages. For the smaller units of the 10-mgd plant, this placement is not required for maintenance; instead, space is provided between the units to allow crane access.

6. 3. 2 Brine Heater System

Reactor coolant, heated to 270°F, flows from the reactor to the four brine heaters. The coolant flows in parallel through the two heaters in the first stage and raises the brine temperature in the first section of the water plant 11°F. It then flows in series through the second- and third-stage heaters and returns to the reactor at 200°F. Four brine heaters provide approximately equal surface area per heater and permit a better arrangement in the brine heater pit. The brine for the other sections of the water plant is also heated 11°F, reaching a temperature of 222°F in the second-stage heater and 201°F in the third-stage heater. By means of positive shutoff butterfly valves at the recycle pumps and at the brine inlets to the hottest stages, the brine flow can be regulated, and brine flow to each brine heater can be shut down independently in the event of trouble (such as a leak in the brine heater) without affecting the operation of the other sections. Brine heater data are listed in Table 6-2.

6. 3. 3 Ejector System

The deaerator stage is evacuated and held at an absolute pressure of 0.7 inches of mercury with a steam jet ejector system which is sized to remove non-condensables at 70°F in approximately the following amounts:

TABLE 6-2

BRINE HEATER DATA - 50-MGD PLANT

FIRST STAGE (Two Parallel Brine Heaters)

LMTD	10.3°F
Duty	5.4 x 10 ⁸ Btu/hr (Total)
Uo	429 Btu/hr ft ² °F
Brine flow	52.3 x 10 ⁶ lb/hr (Total)
Reactor coolant flow	86.6 ft ³ /sec (Total)
Heat transfer area	12.25 x 10 ⁴ ft ² (Total)

Brine Heater Data

Tube velocity	6 fps
Shell velocity	3 fps
Tubes	1-1/4-in. OD, 19 gauge, 70-30 CuNi
Quantity of tubes per heater	3150
Tube length	64.25 ft
Pitch	2.1 in. triangular
Tube sheet diameter	10.3 ft
Baffle spacing (25% segmented)	42.9 in.
Quantity of baffles per heater	18
Pressure drop across shell	4 psi
Pressure drop across tubes	2.5 psi

SECOND AND THIRD STAGES

LMTD	14.6°F
Duty	4.3 x 10 ⁸ Btu/hr
Uo	443 Btu/hr ft ² °F
Brine flow	38.5 x 10 ⁶ lb/hr
Reactor coolant flow	86.6 ft ³ /sec
Heat transfer area	6.64 x 10 ⁴ ft ²

Brine Heater Data

Tube velocity	6.75 fps
Shell velocity	4.60 fps
Tubes	1-1/4-in. OD, 19 gauge, 70-30 CuNi
Quantity of tubes	3120
Tube length	70.6 ft
Pitch	2.35 in. triangular
Tube sheet diameter	11.45 ft
Baffle spacing (25% segmented)	41.8 in.
Quantity of baffles	20
Pressure drop across shell	8.2 psi
Pressure drop across tubes	3.5 psi

Dissolved air	1000 lb/hr
Leakage air	2000 lb/hr
Carbon dioxide	3000 lb/hr

Dissolved air is that entering with the sea water make-up. Leakage air was estimated from behavior of currently operating multistage flash evaporators. The carbon dioxide is that generated by acid treatment of the sea water feed and remaining after degassification in the open tank. It was assumed that the degassing tank is about 50% efficient.

The ejectors have steel bodies with stainless steel nozzles. Ejector condensers have steel shells, 90-10 Cu-Ni tubes and tubesheets, and epoxy-coated steel channels.

A gas-fired packaged boiler system will provide 35,000 lb/hr of steam at 170 psig for operating the ejectors.

6.3.4 Brine Rejection System

The brine blowdown is that portion of the circulating brine which is rejected to the sea. Total dissolved solids in this brine will reach 70,000 ppm. Multistage, vertical, wet-pit pumps fabricated with Ni-Resist cases and impellers of Type 316 stainless steel are used. They are mounted on a deep vacuum sump.

The makeup water must be acidified, degassed, and deaerated before mixing with the recycle brine. It is pumped by a single-stage, vertical, wet-pit pump located in the main sea water inlet structure. This pump is similar to the brine blowdown pump. Motor-operated, vane-type valves, rubber-lined for tight shutoff, serve both as flow control and block valves. The piping immediately after acidification, and also the deaerator vessel including its trays, are lined with neoprene or epoxy plastic. Flow from the degassing tank to the deaerator is controlled by a motor-operated valve, actuated from the level in the deaerator. The degassifier is a 20 by 150-foot rectangular concrete vessel, lined with a bituminous-neoprene composition, mopped-on and air-cured.

6.3.6 Acid Injection System

A vertical steel tank, 32 feet in diameter by 16 feet high, holds two weeks' supply of 98 percent sulfuric acid. This acid can be stored in steel with negligible corrosion. A metering pump with a teflon diaphragm is used for acid injection at a rate of about 4-3/4 gpm. To acidify net makeup water to a pH of 4.0 requires about 125 ppm acid. With this addition, the pH of the recycle stream will be close to 7.0. A pH recorder-controller on the brine stream regulates the stroke of the metering pump. The mixing nozzle for injection will be corrosion resistant.

6.3.7 Recycle Brine Pumps

Multistage, vertical, wet-pit pumps, mounted on a deep vacuum sump, are provided for brine recycle. These are of the same construction as the brine blowdown pumps. Motor-operated, neoprene-lined butterfly valves are used both for throttling control and positive shutoff.

6.3.8 Product Water System

Product water flows via internal loop seals from stage to stage in open troughs beneath the condenser bundles. Individual conductivity elements in each of the external loop seals between vessels provide a way to locate sea water contamination, and can be observed as desired from multipoint indicating instruments in the control room.

Product water, at about 80°F, will collect in a deep vacuum sump, from the trough in the last stage of heat recovery. Product water from the heat rejection stages (about six percent of the total production, at about 69°F) will also flow into the same sump. The product is finally metered, checked for purity by a conductivity meter and by radiation monitors, and is pumped to the plant boundaries for delivery to storage facilities. Delivery pressure head at the site boundary is 800 ft (344 psig).

6.4 PLANT OPTIMIZATION AND SPECIAL PLANT FEATURES

6.4.1 General

Ideally, the cost of nuclear heat should be well established before planning the water plant since the design of the optimum water plant is a function of energy cost. However, the design of the heat source cannot be defined well enough to permit cost estimation until the total thermal energy requirement is determined. Thus, the process of optimization is a stepwise one requiring several iterations.

A cost of $23\text{¢}/10^6$ Btu was chosen for the water plant optimization. This value was taken from report No. COO-278 and was based on a preliminary reactor design concept using aluminum fuel cladding and advanced design technology. At the completion of this study, it appeared that an energy cost of approximately $30\text{¢}/10^6$ Btu was more realistic. This cost is based on the use of Zircaloy cladding and present-day technology. Although this cost is significantly higher than the assumed value, the water plant design, in this case, is determined by practical limitations on the interstage terminal temperature differences and the total number of stages, rather than by a theoretical optimization. At a heat cost of $23\text{¢}/10^6$ Btu, the upper practical limit on performance ratio has been virtually attained and, even with the increase in the cost of heat, the performance ratio cannot be shifted to a significantly higher value. Therefore, the water plant performance ratio and the capital cost will be only slightly increased by an optimization and the effect on the cost of water will be negligible.

The following discussion describes the general rationale and method of optimization followed in developing the reference water plant design.

6.4.2 Plant Optimization Procedure

It is desired to optimize the 50-mgd water plant for converting sea water at a minimum cost when using heat from a deep pool reactor operating between 270°F and 200°F. The problem is complicated by the fact that the brine heaters for the water plant must also act as coolers for the reactor plant; the brine must therefore enter the heaters at some temperature below 200°F if it is to cool the reactor coolant to 200°F. However, in a multistage flash system optimized for 23¢/10⁶ Btu, the brine should be heated only about 10°F above its entering temperature. These requirements suggest that brine heaters, each using an approximate 10°F temperature difference, should be placed in series across the temperature range of the reactor, and that each of these brine heaters should be connected to separate, fully optimized water plants. No other arrangement which will utilize each increment of thermal energy at its proper availability appears to be possible. The optimum number of brine heaters to be used was determined in the following way:

The thermal driving force in a typical optimized stage of the water plant is expected to be about 5.6°F. In an optimized plant, the cost of transferring a Btu will be about the same whether the transfer occurs in the brine heater or in the condenser sections. Therefore, an approximate value for the optimum log-mean temperature difference across the brine heater should be:

$$\frac{(\text{LMTD})_{\text{Heater}}}{\text{Heater}} = \frac{(\$/\text{Heater ft}^2) (U \text{ of condenser})}{(\$/\text{Condenser ft}^2) (U \text{ of heater})} = \frac{(\text{LMTD})_{\text{condenser}}}{\text{condenser}}$$

With a price for brine heater area of \$5.50/ft², for condenser area of \$2.70/ft², and with U's of 443 and 504 respectively,

$$\frac{(\text{LMTD})_{\text{Heater}}}{\text{Heater}} = 13.0^\circ\text{F}$$

Using brine heaters each with an 11°F temperature rise, and each with the above thermal driving force, it is seen that a maximum of three can be placed in series. The hottest section of the plant must operate below or at 250°F under the criteria of this study; by reducing the LMTD from the above value to 10.3°F this section of the plant reaches the limiting temperature, while the other sections operate at the maximum possible temperature consistent with the optimum LMTD and with the physical constraints involved in coupling the water plant to the reactor. Calculated data and a diagram of this arrangement of brine heaters are given in Figure 6-2.

FIGURE 6-2
BRINE HEATER CHARACTERISTICS
50 MGD PLANT

The integration of the three sections of the water plant, each operating at a different top temperature, was done as follows: It was known that at the expected cost for heat at the brine heaters and with the constraint that the cost of adding a flashing stage to the plant is mainly the cost of adding a bulkhead within a vessel, that optimization will occur at a large number of stages, in all probability an impractical number. Therefore, it was desirable to constrain the calculations to a practical temperature drop per stage, selecting this temperature drop in such a way that there are an integral number of stages for each of the three sections of the plant, and so that the same number of heat rejection stages can be used for each of the sections. A mean temperature drop per stage of 4.15°F is such a value; it leads to 43, 36, and 31 stages for the high, intermediate, and low temperature sections respectively, and results in sufficient pressure drop even within the coolest stages to convey the flashing brine from one stage to the next. Accordingly, this mean temperature drop was adopted for the preliminary design.

Using this mean temperature drop per stage, the stage-by-stage description of the hottest plant section was calculated. From this description, the terminal temperature difference (TTD) of the stage following the brine heater was determined for each of the three sections of the water plant. Since the TTD and the number of stages are independent variables in the calculation and since these values were known, the characteristics of the remaining two sections of the plant, the intermediate and low temperature sections, could be found.

Since the curves of water cost versus TTD at a fixed number of stages are very flat in the region of the optimum, and inasmuch as the above plant is in this region (because the TTD resulted from assuming a temperature rise through the brine heater of 11°F which is known from experience to be close to optimum), the conceptual design of the water plant could then be completed upon the assumption of the above TTD's and numbers of stages. The substitution of this plant for a completely optimized facility can therefore be expected to result in a negligible change in the cost of water.

With the above procedure for determining the TTD and the number of stages, the characteristics of any stage within a section are identical, except for capacity, with those of the stages operating at the same temperature in the other two plant sections. A stage operates with the same effectiveness irrespective of the plant section in which it happens to be located. Thus, when the design is optimum for one section of the plant it is optimum for all sections and the plant behaves as a complete unit instead of as three sections. The detailed thermal performance of the high temperature section of this plant is shown in Table 6-3. The table is self-explanatory with the exception that, since the computer program was designed for use with a steam-heated brine heater, it was

necessary to use artificial steam parameters in order to obtain the correct LMTD. Therefore, for this application, the steam properties and overall brine heater heat transfer coefficient shown have no significance.

6.4.3 Special Features Affecting Costs

The following features affect the cost of water but do not depend upon the TTD and the number of stages used in the optimization. In deriving the cost of water, therefore, additional costs for these features must be included.

1. A packaged gas-fired boiler unit for generating approximately 35,000 lb/hr of saturated steam at 185 psia is required. Annual operating charges for this boiler plant, using $35\text{¢}/10^6$ Btu gas at 80 percent efficiency are included in the annual costs of water.
2. There are sufficient sidewalls in the vessels of the three sections of the 50-mgd plant to divide it into four separate units so that each unit, in conjunction with its particular brine heater, may be shut down separately without affecting the others. These sidewalls are of the same thickness carbon steel plate as the remainder of the vessel structures and are not required in a conventional 50-mgd plant.
3. So that each brine heater may be isolated to prevent product contamination in the event of leaks, a butterfly valve, (neoprene-lined for positive shut-off) is included in each brine line, one being at the recycle brine pump for throttling as well as shut-off, and the other being near the brine inlet to the flashing chamber of the hottest stage of the plant section.
4. Although the total pumping capacity and horsepower are about the same as for a conventional 50-mgd plant, more pumps of smaller size are required where different sections of the plant must be isolated. These cost more per unit of water produced than for a single line plant because of the smaller size.
5. There are four brine heaters of special size and construction. These are furnished with double tubesheets to minimize the possibility of cross-leaks between the brine and primary coolant for the reactor, and are of larger than ordinary size because they are liquid-liquid exchangers. For this study the brine is pressurized about 5 psig above the primary coolant so that should leakage occur, the coolant rather than the flashing brine would become contaminated. Costs of maintaining this backpressure on the brine must be included in the annual costs of water.

6. All costs of the sea water intake and outfall structure must be borne by the water plant. At the California site selected such costs must include means for preventing the entry of sea weed and kelp. Such means would range from a simple canal with trash rakes to a large inlet line, extending to sea beyond the kelp beds. The cost of a 1500-foot inlet and discharge lines was included in **this** study.
7. In addition to the customary quality controls for potable water, provision for radiation monitoring must be made in a nuclear-fueled sea water conversion plant, and the annual cost for such testing must be included in the annual cost of water.
8. A discharge head of 800 feet at the site boundary was used in this study to allow for delivery to distribution or storage facilities. This pumping cost was added to the cost of water production.

Section 7

ELECTRICAL SYSTEM

The plant electrical system arrangement is shown on the Electrical Single-Line Diagram, Drawing DA-7.

Electric power for the plant is obtained from two separate sources: (1) a 220-kilovolt line which supplies electric power for the entire water plant and for approximately one-half of the nuclear plant, and (2) a 33-kilovolt line which supplies the other half of the nuclear plant.

Power to the water plant is supplied independently from the nuclear plant through the main water plant transformer. It is estimated that twenty megawatts will be needed to operate the entire water plant.

Normally, the nuclear plant receives power from both the 220-kv and 33-kv lines. The incoming voltages are reduced to 4160 volts through the main nuclear plant transformer and the reserve nuclear plant transformer. The main coolant pumps and the load center transformers are supplied from two 4160-volt busses. The load center transformers step down the voltage to 480 volts.

The 480-volt system is used to operate all reactor auxiliary equipment and to supply electric power to other loads in the reactor plant and auxiliary buildings. The 480-volt system consists of two busses supplying non-essential loads and two essential load centers supplying the loads required for safe short-term operation and orderly shutdown of the reactor. The essential load centers are backed up by an emergency diesel generator set that starts automatically in the event that incoming power from either the 200-kv line or the 33-kv line is interrupted. At the same time, the interrupted 4160-volt bus can be manually switched to the operative incoming power supply so that both main coolant pumps can continue to operate, thus allowing normal nuclear plant operation from one power source. Since the water plant cannot be operated from the 33-kv power source, loss of 220-kv power will require water plant shutdown. When the water plant is shut down, eliminating the reactor heat removal system, the reactor must be scrammed. If the 33-kv power source is available, provision is made for pumping cold sea water to the third stage brine heater. Thus, even though the reactor must be shut down on loss of the 220-kv power source, a normal cooldown can be made by using the main coolant pumps with the third stage brine heater as a heat removal system.

If the 33-kv power supply is interrupted, both the water plant and nuclear plant can continue to operate normally with backup power for the nuclear plant available from the diesel generator. If both the 220-kv and 33-kv

power sources fail, the diesel generator will supply power to the essential load centers, allowing a safe shutdown of the reactor.

Three 208/120-volt AC instrument busses are provided. Two are normally supplied from the 480-volt essential load centers through instrument MG sets. Instruments on these busses are isolated from transients in the 480-volt supply by the MG sets. In addition, on complete loss of AC power, these busses can be powered by the battery through the MG sets. The third 208/120-volt AC instrument bus is supplied through either the No. 1 or No. 2 instrument transformer. Normally the No. 1 instrument transformer is on the line. In the event of a failure in the 220-kv line, the bus is automatically switched to instrument transformer No. 2 which is powered by the 33-kv line.

During normal operation the 125-volt DC system is supplied from both MG sets which are permanently on the line. In case of complete AC system failure in the plant, the battery will supply all necessary DC power to operate the reactor control system and nuclear instrumentation for orderly reactor shutdown.

NOTE: FLOW DIAGRAM TYPICAL
FOR ALL PLANTS.
DATA SHOWN FOR 23.2 MGD
SECTION OF 50 MGD PLANT.

FLOW DIAGRAM

MULTI-STAGE FLASH EVAPORATORS

BFGHTEL CORPORATION

SCIENTIFIC DEVELOPMENT DEPT.

301 MISSION STREET, SAN FRANCISCO CALIFORNIA

SALINE WATER CONVERSION PLANT OPTIMIZATION

CHICAGO OPERATIONS OFFICE 23.2MGD PLANT 255F STEAM 60F SINK

PRODUCTION= 8050000. LBS/HR. RECYCLE RATIO= 6.49 LBS/LB PRODUCT

CONCENTRATION RATIO= 1.70, TOTAL NUMBER OF STAGES IS 43

TEMPERATURE OF FLASHING BRINE FROM BRINE HEATER IS 249.50 DEG F.

SEA WATER TEMPERATURE IS 60.00 DEG F. APPROXIMATE ITD# 4.15 DEG F.

STAGE NO.	FLASHING BRINE, MLB/HR	DISTILLATE PRODD, MLB/HR	RECYCLE BRINE, MLB/HR	CONCENTRATION WT. PERCENT	PRODUCTION M LB/HR	PRESSURE PSIA	VAPOR FLOW M CU FT/SEC	BOILING POINT ELEV. DEG F.
1	52015.	221.	52236.	5.9753	221.	26.2146	981.12	2.5847
2	51795.	441.	52236.	6.0007	220.	24.3017	1050.28	2.5732
3	51576.	660.	52236.	6.0261	219.	22.5010	1125.51	2.5615
4	51359.	877.	52236.	6.0516	217.	20.8081	1207.41	2.5497
5	51144.	1092.	52236.	6.0771	216.	19.2186	1296.69	2.5376
6	50929.	1307.	52236.	6.1027	214.	17.7282	1394.09	2.5254
7	50716.	1520.	52236.	6.1283	213.	16.3324	1500.49	2.5130
8	50505.	1731.	52236.	6.1540	211.	15.0772	1616.81	2.5004
9	50295.	1941.	52236.	6.1797	210.	13.8081	1744.13	2.4877
10	50086.	2150.	52236.	6.2054	209.	12.6713	1883.62	2.4748
11	49879.	2357.	52236.	6.2312	207.	11.6125	2036.62	2.4618
12	49673.	2563.	52236.	6.2570	206.	10.6274	2204.68	2.4486
13	49469.	2767.	52236.	6.2828	204.	9.7130	2389.33	2.4353
14	49267.	2969.	52236.	6.3086	203.	8.8651	2592.47	2.4218
15	49066.	3170.	52236.	6.3344	201.	8.0801	2816.19	2.4082
16	48866.	3370.	52236.	6.3603	199.	7.3544	3062.82	2.3945
17	48669.	3567.	52236.	6.3861	198.	6.6845	3334.98	2.3807
18	48472.	3764.	52236.	6.4120	196.	6.0672	3635.59	2.3667
19	48278.	3958.	52236.	6.4378	195.	5.4991	3967.96	2.3527
20	48085.	4151.	52236.	6.4636	193.	4.9771	4335.79	2.3386
21	47894.	4342.	52236.	6.4894	191.	4.4984	4743.21	2.3243
22	47704.	4532.	52236.	6.5152	189.	4.0599	5194.90	2.3100
23	47517.	4719.	52236.	6.5410	188.	3.6591	5696.09	2.2957
24	47331.	4905.	52236.	6.5667	186.	3.2932	6252.66	2.2812
25	47146.	5090.	52236.	6.5923	184.	2.9597	6871.19	2.2667
26	46964.	5272.	52236.	6.6179	182.	2.6563	7559.09	2.2522
27	46783.	5453.	52236.	6.6435	181.	2.3808	8324.64	2.2377
28	46605.	5631.	52236.	6.6689	179.	2.1309	9177.15	2.2231
29	46428.	5808.	52236.	6.6943	177.	1.9048	10127.00	2.2085
30	46253.	5983.	52236.	6.7196	175.	1.7003	11185.83	2.1939
31	46080.	6156.	52236.	6.7448	173.	1.5159	12366.60	2.1793
32	45910.	6326.	52236.	6.7699	171.	1.3498	13683.77	2.1647
33	45741.	6495.	52236.	6.7949	169.	1.2005	15153.41	2.1502
34	45574.	6662.	52236.	6.8197	167.	1.0664	16793.34	2.1357
35	45410.	6826.	52236.	6.8444	165.	0.9462	18623.26	2.1212
36	45247.	6989.	52236.	6.8690	162.	0.8387	20664.90	2.1069
37	45087.	7149.	52236.	6.8934	160.	0.7426	22942.06	2.0925
38	44930.	7307.	52236.	6.9176	158.	0.6569	25480.74	2.0783
39	44774.	7462.	52236.	6.9416	155.	0.5806	28309.10	2.0642
40	44621.	7615.	52236.	6.9654	153.	0.5128	31457.50	2.0502

HEAT REJECTION SECTION

41	44475.	7761.	50910.	6.9883	146.	0.4544	33726.04	2.0367
42	44330.	7906.	50910.	7.0112	145.	0.4018	37809.65	2.0232
43	44186.	8050.	50910.	5.9500	144.	0.3547	42434.80	2.0097

STAGE NO.	TEMP FLASHING BRINE, DEG F	TEMP DIST PRODD, DEG F	TEMP RECYCLE BRINE, DEG F	TERMINAL TEMP DIFFERENCE, F	LOG MEAN TEMP DIFFERENCE, F	HEAT TRANSFER MBTU/STAGE	OVERALL U BTU/FT2/F/HR	AREA SQ FT/STAGE
1	245.30	242.71	238.56	4.1500	6.0159	210236.	648.85	53859.
2	241.08	238.51	234.34	4.1692	6.0397	210504.	647.03	53867.
3	236.86	234.30	230.11	4.1887	6.0634	210744.	645.12	53876.
4	232.63	230.08	225.87	4.2085	6.0872	210958.	643.12	53887.
5	228.39	225.85	221.62	4.2286	6.1110	211143.	641.02	53900.
6	224.14	221.62	217.37	4.2490	6.1350	211300.	638.83	53914.
7	219.89	217.38	213.11	4.2698	6.1590	211427.	636.54	53929.
8	215.63	213.13	208.84	4.2911	6.1832	211523.	634.15	53944.
9	211.37	208.88	204.57	4.3128	6.2077	211588.	631.66	53961.
10	207.10	204.62	200.29	4.3351	6.2323	211621.	629.06	53979.
11	202.83	200.37	196.01	4.3579	6.2572	211620.	626.34	53997.
12	198.56	196.11	191.73	4.3814	6.2823	211585.	623.52	54015.
13	194.28	191.85	187.44	4.4054	6.3078	211513.	620.58	54034.
14	190.01	187.58	183.15	4.4301	6.3336	211405.	617.51	54053.
15	185.73	183.32	178.87	4.4555	6.3598	211258.	614.33	54072.
16	181.46	179.06	174.58	4.4817	6.3863	211071.	611.02	54090.
17	177.18	174.80	170.29	4.5086	6.4132	210842.	607.59	54109.
18	172.92	170.55	166.01	4.5364	6.4406	210570.	604.02	54128.
19	168.65	166.30	161.73	4.5650	6.4684	210253.	600.32	54146.
20	164.39	162.05	157.46	4.5944	6.4967	209889.	596.48	54163.
21	160.14	157.82	153.19	4.6248	6.5255	209477.	592.50	54180.
22	155.90	153.59	148.93	4.6562	6.5547	209014.	588.37	54196.
23	151.66	149.36	144.68	4.6886	6.5845	208498.	584.10	54212.
24	147.43	145.15	140.43	4.7219	6.6148	207928.	579.68	54226.
25	143.22	140.95	136.20	4.7564	6.6456	207300.	575.10	54239.
26	139.02	136.77	131.97	4.7919	6.6770	206612.	570.37	54252.
27	134.83	132.59	127.76	4.8286	6.7090	205863.	565.48	54263.
28	130.66	128.43	123.57	4.8664	6.7416	205049.	560.42	54272.
29	126.50	124.29	119.39	4.9055	6.7747	204168.	555.20	54281.
30	122.37	120.17	115.23	4.9457	6.8085	203218.	549.81	54287.
31	118.25	116.07	111.08	4.9873	6.8429	202194.	544.24	54292.
32	114.16	111.99	106.96	5.0302	6.8779	201095.	538.50	54295.
33	110.08	107.93	102.86	5.0744	6.9136	199917.	532.57	54296.
34	106.04	103.90	98.78	5.1201	6.9499	198657.	526.46	54295.
35	102.02	99.90	94.73	5.1672	6.9869	197312.	520.16	54291.
36	98.03	95.93	90.71	5.2158	7.0245	195877.	513.67	54285.
37	94.07	91.98	86.72	5.2659	7.0629	194350.	506.98	54276.
38	90.15	88.07	82.75	5.3176	7.1020	192726.	500.09	54265.
39	86.26	84.19	78.82	5.3711	7.1418	191001.	492.98	54250.
40	82.41	80.36	74.93	5.4262	7.1823	189170.	485.67	54231.
TOTAL AREA=								2165106.
41	78.71	76.68	71.07	5.6035	7.2955	181298.	511.62	48573.
42	75.02	73.00	67.38	5.6190	7.3115	181350.	503.96	49217.
43	71.33	69.32	63.69	5.6368	7.3287	181272.	495.98	49870.
TOTAL AREA=								147660.
GRAND TOTAL AREA=								2312766.

BRINE HEATER DATA

STEAM PRESSURE= 32.7 PSIA, STEAM TEMPERATURE= 255.3 DEG F.

TEMPERATURE RISE OF BRINE THROUGH HEATER= 10.939 DEG F. LOG MEAN TEMPERATURE DIFFERENCE= 10.327

OVERALL HEAT TRANSFER COEFFICIENT = 584.69 BTU/FT2/F/HR

HEAT REQUIREMENT OF HEATER IS 545.569 MBTU/HR, AREA REQUIREMENT OF HEATER IS 90357. SQ.FT.

STEAM REQUIREMENT= 579220. LB/HR

PERFORMANCE RATIO= 14.76 LBS PRODUCT/MBTU

DESIGN DATA

HEAT RECOVERY SECTION

NUMBER OF TUBES PER STAGE IS 27567.
VELOCITY OF RECYCLE BRINE IN TUBES IS 5.569 FT/SEC
PRESSURE DROP THROUGH THIS SECTION IS 53.06 PSI
LENGTH OF TUBES IS 12.00 FEET
OUTSIDE DIAMETER OF TUBES IS 0.6250 INCHES
WALL THICKNESS OF TUBES IS 0.0490 INCHES
TOTAL WEIGHT OF TUBES IN EACH STAGE IS 108801. POUNDS

HEAT REJECTION SECTION

NUMBER OF TUBES PER STAGE IS 21373.
VELOCITY OF RAW SEA WATER IN TUBES IS 7.000 FT/SEC
PRESSURE DROP THROUGH THIS SECTION IS 8.42 PSI
LENGTH OF TUBES IS 14.07 FEET
OUTSIDE DIAMETER OF TUBES IS 0.6250 INCHES
WALL THICKNESS OF TUBES IS 0.0490 INCHES
TOTAL WEIGHT OF TUBES IN EACH STAGE IS 98937. POUNDS

BRINE HEATER

TOTAL NUMBER OF TUBES IN HEATER IS 5226.
VELOCITY OF RECYCLE BRINE IN TUBES IS 6.000 FT/SEC
PRESSURE DROP THROUGH THIS SECTION IS 15.65 PSI
LENGTH OF TUBES IS 52.83 FEET
OUTSIDE DIAMETER OF TUBES IS 1.2500 INCHES
WALL THICKNESS OF TUBES IS 0.0420 INCHES
TOTAL WEIGHT OF TUBES IN BRINE HEATER IS 163247. POUNDS

TOTAL RECYCLE BRINE PUMP HEAD IS 94.57 PSI

Section 8

SAFETY CONSIDERATIONS

8.1 GENERAL

The inherent safety of the deep-pool reactor concept is one of its principal attractions. During normal operation, discharges of radioactivity to the environment can be kept to negligible levels and, as with any power reactor, these normal discharges will have no effect on the environment and will not limit the operation of the plant at any location. For the most severe accidents which can be assumed, the amount of radioactivity released from the reactor is also small and can be effectively held up in the plant with relatively simple safeguards systems. Even if all of the radioactivity which could be released from the reactor during the most severe accident considered were released directly to the environment, the consequences to the public would be slight.

Throughout the study, an effort has been made to identify areas having safety significance and, where appropriate, features were incorporated into the plant design to improve its overall safety. It is recognized that unanticipated potential accident situations may be identified during the detailed design and safety analysis of the plant. However, it is significant that all of the accidents which were identified in the preliminary safety evaluation could be coped with easily by the inherent features of the concept or by relatively minor design changes. Because of the inherent safety features of this plant, it is expected that other potential accidents could be handled in a similar manner without altering the basic concept or unduly complicating the design, and without increasing the risk to the public.

This section discusses the radioactivity and radiation levels of significance during normal operation and presents the results of the preliminary safety evaluation made during the study. The possible consequences of several assumed accidents are discussed.

8.2 NORMAL OPERATION

8.2.1 Reactor Coolant Activity

The reactor coolant is the principal source of radioactive liquid waste discharged from the plant. The short-lived coolant activity establishes the requirements for shielding the main coolant system during operation. The sources and estimated levels of coolant activity are presented in the following paragraphs.

Coolant Activation

During transit through the core, the coolant becomes radioactive due to activation of various nuclides in the coolant. Of the activation products produced, the most important are A-41, N-16, N-17 and corrosion products. The A-41 is a product of the argon which is dissolved in the coolant during initial fill and as a result of air in-leakage. The nitrogen species are products of O-16 and O-17 in the coolant.

The following values for the N-16 and N-17 activities at the core exit were estimated.

<u>Species</u>	<u>Activity</u>
N-16	300 $\mu\text{c/ml}$
N-17	$10^5 \text{ n/cm}^2 \text{ sec}$

It can be assumed that the N-16 and N-17 activity levels at the top of the reactor tank are negligible, since their half-lives are short (about seven and four seconds, respectively) and since transport can be accomplished only by convection currents which in this case are quite small. This allows more than sufficient time for decay of N-16 and N-17 to insignificant levels.

The main coolant system is designed to prevent in-leakage of air. In addition to minimizing corrosion, this reduces the amount of argon in the coolant and reduces the production of A-41. However, some air is always present in the coolant, the principal source being that initially contained in the water prior to reactor operation. If the deaeration system removes 99 percent of the air and argon dissolved in the coolant, the A-41 activity in the coolant would be about 0.05 $\mu\text{c/ml}$.

Corrosion Products

The principal long-lived isotopes expected to contribute to the main coolant activity are those due to activation of corrosion products. Fission products, discussed in the following paragraphs, may be the major source of activity under abnormal conditions. Due to the unique features of the system, it is difficult to calculate with any degree of accuracy the activity level due to activated corrosion products ("crud"). However, an order-of-magnitude estimate can be obtained by recognizing that the reactor coolant purification system has approximately the same design characteristics as those of other power reactors and should maintain approximately the same crud levels. The ion exchangers are expected to remove approximately 99 percent of all corrosion products based on the assumed corrosion rates discussed in Section 5.

Based on data from the Indian Point reactor, ⁽¹⁸⁾ the calculated corrosion product activity would be on the order of 0.2 $\mu\text{c}/\text{ml}$. This is only an order-of-magnitude estimate since the activity level depends on many interrelated variables such as fuel cladding material and temperature, and amounts, types, and corrosion rates of materials in the main coolant system. Representative crud specific activity levels for several PWR's have been reported to be on the order of 10^7 dis/min/mg of material, (approximately $3 \times 10^2 \mu\text{c}/\text{mg}$). Since the coolant purification system is designed to maintain a crud level of one ppm, or 10^{-3} mg/ml, the calculated activity level is 0.3 $\mu\text{c}/\text{ml}$, which is in agreement with the prior value. Since the reactor system is carbon steel, with a predominant proportion of the surface area being cupro-nickel, it is not representative of reactor systems for which data are available. However, the purification system is designed to accommodate the corrosion rates expected in the system, so that if the specific activity of the crud produced is of the same order of magnitude as that from other plants, the crud activity in the coolant should be of the same order of magnitude also.

Surface Contamination of Cladding

Fission product activity may be released directly to the main coolant from tramp uranium associated with the fuel cladding. Depending on the location of the uranium atoms within the cladding structure, the fission products will contaminate the primary water either by recoiling directly into the water or by diffusion through the cladding matrix. Using the assumption that the fuel cladding surfaces will be contaminated at a level of 5 micrograms of U-235 per square foot, the water activity due to surface contamination can be estimated by comparison with estimates for the LaCrosse boiling water reactor (LACBWR)⁽¹⁹⁾. Adjusting these data for the relative core surface areas, operating power levels and main coolant volumes, the main coolant activity concentration due to cladding contamination is estimated to be about $3 \times 10^{-2} \mu\text{c}/\text{ml}$ for this reactor.

Fuel Pin Defects

Fission products could be released to the main coolant due to defects in the fuel element cladding. Fission product buildup in the coolant due to the presence of defects in 10 pins can be estimated, by comparison to the analysis made for the LACBWR reactor, to be approximately 0.001 $\mu\text{c}/\text{ml}$.

In summary, estimated values for the various contributions to the coolant activity at the top of the reactor tank are listed in Table 8-1. At the reactor outlet the activity level is approximately 300 $\mu\text{c}/\text{ml}$ due almost entirely to N-16.

TABLE 8-1

MAIN COOLANT ACTIVITY LEVELS AT TOP OF REACTOR TANK

<u>Source</u>	<u>Activity Levels ($\mu\text{c/ml}$)</u>
A-41	0.05
Corrosion products	0.3
Fission products(cladding tramp U)	0.03
Fission products (fuel defects)	<u>0.001</u>
TOTAL	0.38

8.2.2 Spent Fuel Pit Water Activity

Activity of the spent fuel pit water will be due primarily to fuel pin defects and the associated fission product release. The release rate per pin defect should be much smaller in the spent fuel pit than it is for a pin undergoing irradiation in the reactor for several reasons. First, decaying pins will be much cooler than those being irradiated. Diffusion rates are proportional to the exponential of the temperature and the fission product diffusion rates will be negligible for cool pins. Second, recoil, a process which can account for a sizable fraction of the fission product release in the reactor, is non-existent in decaying pins. Leaching is perhaps the most significant mechanism for release of fission products in the spent fuel pit but this should be a small effect also.

Comparing the available activity sources for the spent fuel pit with those for the main coolant, as given in Table 8-1, it is expected that the activity in the spent fuel pit water will be at worst a factor of 400 less than that in the reactor coolant at the top of the reactor tank during operation, or about 0.001 $\mu\text{c/ml}$.

8.2.3 Radiation Levels

Reactor Pit

The radiation level at the top of the reactor pit is due primarily to activated corrosion products in the main coolant. It is assumed that adequate shielding is provided in the annulus between the reactor tank and the reactor pit wall to reduce radiation streaming from the bottom of the tank to negligible levels. Using the estimated activity levels shown in Table 8-1, and assuming an average gamma energy per disintegration of 2 Mev, the radiation level two feet above the top of the reactor tank

would be approximately 450 mr/hr. The steel provided by the reactor tank lid and the reactor pit cap (approximately four inches combined thickness) reduces the radiation level to approximately 10 mr/hr.

Brine Heater Pit

Radiation levels at the top of the brine heater pit during operation are due almost entirely to the short-lived isotope N-16. Using the calculated activity level of N-16 at the core exit and the physical dimensions of the brine heaters and brine pit, the radiation level at the top of the pit during operation is estimated to be 25 R/hr. The four feet of concrete shielding over the pit reduces the radiation level to an acceptable value. Due to the more rapid decay of N-17, its contribution to the radiation level at the pit top is negligible.

8.2.4 Gaseous Activity Release

Fission Product Gases

During normal operation, a small fraction of the generated fission products may escape from the fuel elements into the primary coolant, principally from fuel pin defects. The estimated fission gas release rate is based upon the criterion that on the average, ten fuel pins (approximately 0.2 percent of the total) will contain defects and release 10 percent of the fission gas generated internally. This criterion was provided by COO as a basis for the study.

Of primary interest are the isotopes Xe-133, Xe-135, Kr-85, Kr-87, and Kr-88, with a combined total effective fission yield of approximately 0.2 atoms/fission. At a power level of 400 Mwt, the estimated release rate of gases is 1×10^{14} atoms per sec, or 1.5×10^{-5} g-mols per day. The resulting activity release rates, expressed in microcuries per second, are shown in Table 8-2.

TABLE 8-2

FISSION GAS RELEASE RATE

<u>Isotope</u>	<u>Release Rate ($\mu\text{c/sec}$)</u>
Xe-133	2.0×10^3
Xe-135	2.5×10^3
Kr-85	1.3
Kr-87	8.5×10^4
Kr-88	5.1×10^4

The planned mode of operation for the radioactive gas handling system will allow a decay period of at least 60 days prior to atmospheric release. Table 8-3 shows the activity release rate after a 60-day decay.

TABLE 8-3
FISSION GAS RELEASE RATE WITH 60-DAY DECAY

<u>Isotope</u>	<u>Release Rate ($\mu\text{c}/\text{sec}$)</u>
Xe-133	2.0×10^{-2}
Xe-135	-
Kr-85	1.3
Kr-87	-
Kr-88	-

After 60 days, all gases except Kr-85 and Xe-133 have decayed to insignificance.

Argon - 41

The radioactive gas A-41 is formed and released to the atmosphere from two sources. The first arises from the air dissolved in the reactor coolant and irradiated by the core. The second is from the ventilation air circulated through the annular space immediately outside the reactor tank.

During operation a certain percentage of the argon in the coolant escapes and is released into the space at the top of the reactor tank. This gas is purged along with the fission gases and stored in the radwaste system. The A-41 activity level throughout the primary coolant is estimated to be $0.05 \mu\text{c}/\text{ml}$ and the total A-41 activity in the coolant is approximately 50 curies. If this total amount of A-41 is released to the radwaste system and held up for 60 days prior to atmospheric release, the A-41 activity will have decayed to a negligible level.

The second source of A-41 is due to neutron activation of the argon in the reactor pit ventilating air as it passes near the core. This A-41 is carried to the stack with the ventilating air. The combined lead-water shield within the reactor tank reduces the neutron flux outside the reactor tank to a level such that the A-41 activity is less than the maximum permissible concentration for unrestricted access when the air leaves the reactor pit.

Leakage of Radioactive Gases

During operation, gases released from the main coolant accumulate in the space at the top of the reactor tank. The rate of increase of partial pressure of these gases within this space is very low and should be sufficiently small to permit only occasional purging of the gases to the rad-waste system.

Some diffusion of fission gases through the tank top seal might occur. However, considering that the tank pressure is below atmospheric when the gas space is being purged, the amount of radioactive gas leakage is expected to be very small. Any leakage that might occur would be carried by the reactor pit ventilating air to the stack. Dilution of this gas with the ventilation air, as well as normal atmospheric dilution, will reduce its concentration at off-site areas to insignificant levels.

8.2.5 Contamination of Product Water

Since reactor coolant and brine are separated by a single barrier in the brine heaters and since there is a finite probability of brine heater tube leakage or failure, there could be some concern that the product water could become contaminated with radioactivity. To minimize this possibility at this stage of the design, several protective features have been incorporated into the plant design.

The pressure of the brine is maintained above that of the reactor coolant in the brine heaters to ensure that a tube failure will result in leakage of brine into the reactor coolant rather than contamination of the brine. Such a failure would be quickly detected by conductivity cells located in the reactor coolant system at the outlet from each brine heater. Brine flow to the faulty heater would be stopped and the heater would be drained and flushed.

The brine heaters are provided with double tubesheets so that any leakage at the tube-to-tubesheet joints would not result in contamination of either the brine or the coolant. Since most leakage would be expected to occur at the tubesheet joints, the double tubesheets should greatly reduce the probability of brine heater leakage.

A radiation monitor is provided at the product water sump as a final check that no contamination has occurred.

Further analysis may indicate that it is not necessary to maintain the brine at a higher pressure than the coolant.

8.2.6 Discharge of Radioactive Liquids

Radioactive liquids are discharged from the plant to the liquid radwaste system for monitoring, temporary storage, treatment and disposal. These wastes are released only after the activity levels have been measured and found to be sufficiently small.

All liquid wastes are released to the discharge from the water plant where they are diluted with return brine and cooling water flowing at a rate of approximately 300,000 gpm. The discharge is carried by marine pipeline approximately 1500 feet off shore. To contaminate this water to the MPC for unknown radioisotopes (10^{-8} $\mu\text{c/ml}$), the rate of activity release would have to be 10 $\mu\text{c/min}$.

Nearly all of the radwaste is primary coolant, which is estimated to have a maximum activity level of 0.4 $\mu\text{c/ml}$. (Refer to Table 8-1.) Liquid waste at this activity level passes through the liquid radwaste system demineralizer in which at least 99 percent of its activity is removed. The allowable release rate of liquid radwaste based on these conditions is on the order of 750 gallons per day. Normal sources of radioactive liquid wastes include pump seal leakage, ground water seepage into the reactor pit, and condensate from the off-gas condenser. These sources are expected to total much less than 750 gallons per day.

If a situation arose in which the rate of liquid radwaste production rose above that which could be directly processed and released, then use could be made of one or both of the 100,000-gallon holdup tanks. This large reserve holdup capacity, coupled with the ability to make use of the main coolant purification system demineralizers in an abnormal situation, provide a high degree of operating flexibility so that the plant can continue to operate for a considerable period of time with generation of radioactive liquids at rates above those normally expected.

8.3 ACCIDENTS

8.3.1 Coolant System Rupture

The design of the reactor plant is such that a major rupture of the main coolant system does not lead to excessive release of radioactivity to the atmosphere. The brine heater pit and reactor pit diameters are such that if a major rupture does occur somewhere in the system and empties the reactor coolant into the pits, the equilibrium water level would be above the top of the core. Subsequent reduction of this level would arise from boiling off some of the coolant due to decay heat production. Maximum boil-off rate would be approximately 200 gpm. An emergency fill line is provided to replenish water lost by boiling, and if necessary the pits could be flooded with sea water.

If the walls of the pit are ruptured, ground water would drain into the pit and keep the core covered since the reactor is located well below the water table. It is expected that this would be the case at any site at which a water desalting plant might be located.

Transient Stage

The system is designed to provide for the release of the flashing reactor coolant without overpressurizing the pits or buildings. The maximum water vapor flow rate is estimated by considering the rupture to occur at the reactor outlet where the coolant temperature is maximum (270°F). If the break is assumed to be the complete rupture of a 40-inch line, then the total coolant flow from both ends of the break is estimated to be 5,000 lb/sec. The corresponding flashing rate is 300 lb/sec, or 8000 ft³/sec of saturated steam. A large relief duct from the brine heater pit to the stack is provided to accommodate this flow rate without overpressurizing the pits and lifting their caps. It is estimated that an approximate total of 31,000 lb of coolant will flash (out of 2.5 x 10⁶ lb) in the event of major rupture. This is roughly equivalent to 1,000,000 cubic feet of saturated water vapor at 212°F.

Activity Release

In a coolant system rupture accident, sources of activity could be: (1) fission products contained in the reactor coolant, (2) corrosion products and (3) activated reactor coolant. Paragraph 8.4 indicates the off-site doses resulting from release to the atmosphere of the activity in the flashing coolant.

8.3.2 Power Failure Accidents

Power is normally supplied by two separate sources, 220 and 33 kv. The 220-kv source supplies the entire water plant and approximately one-half of the nuclear plant. The 33-kv source supplies the other half of the nuclear plant. Failure of the 33-kv source would not necessitate shutdown of either plant since the 200-kv source can supply the water plant and the entire nuclear plant. Power failure accidents fall into two categories: (1) loss of the 220-kv source, and (2) loss of both the 220-kv and 33-kv power sources.

Loss of 220-KV Power Source

If the 220-kv power source is lost, the water plant is shut down, removing the heat sink for the nuclear plant. In this situation, the reactor is automatically scrammed. Coolant flow is maintained by the main coolant pump which is normally supplied by the 33-kv power source. The bypass line, between the main coolant pump discharge and the third stage brine heater automatically opens, diverting coolant flow directly to the

third-stage brine heater. At the same time, valves open in the sea water cooling system allowing cold sea water to be pumped through the brine side of the third-stage brine heater. Since power remains available to the main coolant pumps, a normal cooldown can be accomplished using the third-stage brine heater as a heat sink and no damage or release of activity results.

Loss of Both 220-KV and 33-KV Power Sources

The reactor control system is designed so that upon loss of power, the reactor is immediately shut down. The reactor power drops almost instantaneously to approximately 20 mw from decay heat and continues to decrease more slowly thereafter.

The main coolant pumps are provided with sufficient flywheel energy to maintain coolant flow for approximately one minute. (Refer to Paragraph 5.2.1.) This supplies flow necessary to prevent overheating the fuel immediately following cessation of power.

The large volume of water in the reactor tank provides a "thermal flywheel" for the reactor system; i. e., the decay heat generated in the core is initially absorbed by the water in the tank. It takes on the order of 15 to 20 minutes to heat this water to 212°F, at which time surface boiling would ensue.

During the coastdown period, the bypass line from the main coolant pump discharge to the entrance to the third stage brine heater is opened, diverting flow from all but the last brine heater. Consequently, the hot water in the first three heaters is not pumped back to the reactor tank, and the average temperature in the tank after the main coolant pumps have completely stopped is between 200°F and 205°F.

During the period when decay heat is raising the coolant temperature, the emergency diesel generator is started. It then supplies emergency power to the decay heat removal pump and the sea water cooling pumps. Reactor coolant is then pumped through the decay heat removal system, and sufficient heat is removed to prevent the primary coolant from boiling.

From the above discussion, it is seen that no excess radioactivity would be released to the atmosphere during a loss of power accident. Abnormal amounts of activity would be released only if loss of both the 220-kv and 33-kv power sources is followed by failure of the auxiliary diesel generator. If this triple failure were to occur, boiling of the reactor coolant in the tank would ensue approximately fifteen minutes after loss of power. The rate of boil-off would be on the order of 50 gpm. This could be made up if necessary using a fire hose and hand pump. Even in this case where no power is available, decay heat is removed by

boiling of the coolant at the surface of the tank and no gross fuel damage would occur. The steam produced and any activity entrained in it would be released through the off-gas line to the stack, even if small amounts of steam leak through the reactor tank lid and the pit cap to the building, the total activity release would be small.

8.3.3 Reactivity Insertion Accidents

In general, the category of reactivity insertion accidents consists of those accidents that result from abnormal operation or failure of equipment associated with the control rods and for variation in core coolant temperature other than failure of integrity of the pool tank.

Startup Accident

The startup accident is defined as the unrestricted withdrawal of control rods from the core until the flux level reaches the high flux trip limit and causes the control rods to scram. In order for the accident to occur, it is necessary that the operator continue to withdraw rods without being alerted to the rise in reactor power. In addition, the period trip instrument channels must fail to scram the reactor. Thus, the accident is postulated to occur as a result of two failures and terminated by a scram of control rods which is initiated by the trip from the high flux instrumentation.

The startup accident would cause no significant core damage, since the rod withdrawal rate can be limited so that fuel center temperatures would be below the melting point; heat fluxes generated would be less than those required for clad burnout.

Control Rod Withdrawal at Power

The unrestricted withdrawal of control rods while the reactor is at power would normally be terminated by a scram either from the high flux or period trip instrumentation. As with the startup accident, the control rod withdrawal rate would be limited so that gross fuel damage could not result even if the accident occurred.

For the accident to occur, it is necessary that both the high flux level and period instrumentation fail to scram the reactor. Were this to occur, the temperature of the primary coolant at the outlet would rise. However, since the core outlet coolant temperature is monitored and tied into the reactor scram instrumentation, a scram would occur before the outlet temperature reaches its saturation point. In the event that both the high flux level and the high outlet temperature trips failed, the reactor would continue to increase its power output until core boiling began. With a large negative void coefficient, the core could sustain such operation without damage.

Step-type Reactivity Insertion Accidents

Foremost in consideration of step-type reactivity accidents is the forceful ejection of a control rod when the reactor is already critical. Because of the limited amount of reactivity in each rod and because of the length of the rods, a sizable portion of a rod would have to be rapidly removed in order to simulate a step input of reactivity. To rapidly remove a rod would require some amount of energy to be stored in the drive system. Energy storage of this magnitude is not possible with the rod drive system selected.

Other types of step insertions of reactivity likewise do not appear to be readily achievable. Burnable poison will constitute an integral part of the core and will not be removable from the core during operation.

Cold Water Accidents

The possibility of suddenly inserting cold water into the reactor core is of potential concern because of the negative reactivity coefficient of moderator temperature. However, the magnitude of the cold water accident possible for this reactor is small because of the small change in water density between operating temperature and room temperature. The design of the main coolant system effectively prevents the accumulation of cold water in a form readily available for accidental insertion. Natural circulation within the reactor tank maintains the bulk of the water above the core at a near-uniform temperature of 200°F, the normal core inlet temperature. This large reservoir is an inherent barrier against any sudden introduction of cold water into the core.

8.3.4 Non-reactor Accidents

Other accidents, which are not related to the basic reactor concept, could release radioactivity to the environment. These include fuel cask handling accidents, rupture of radioactive gas holdup tanks and spent fuel pit accidents. Since these accidents cannot be more severe (and in some cases are probably less severe) in the deep-pool reactor concept than for other plants, they have not been considered further in this evaluation.

8.4 ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

8.4.1 General

An evaluation of the consequences of assumed reactor accidents is necessary for all reactors because of the possible severe consequences to the public if a significant fraction of the fission products contained within the fuel were released to the environment. In such analyses it is necessary to consider the magnitude of the possible release of activity from

the core, the amount of activity released from the reactor building to the atmosphere, the degree of atmospheric dilution which would occur during various meteorological conditions, and the distance and direction to inhabited areas. Such considerations form the basis for determining the relative safety of reactor plant designs and provide guides for reactor plant siting.

Of the accidents discussed above, rupture of the main coolant system provides the greatest potential for release of radioactivity to the environment. This accident is therefore considered the reference maximum accident for determining off-site doses under accident conditions. Rupture of a main coolant pipe is also considered to be the maximum accident in many other water-cooled reactors, but in these cases it leads to loss of reactor coolant and often to fuel melting and release of substantial quantities of fission products. As indicated in paragraph 8.3.1, rupture of the main coolant system in the deep-pool reactor will not result in fuel melting since the core will remain covered.

Since the limiting off-site dose as a result of a reactor accident is generally considered to be the dose to the thyroid from inhalation of radioactive iodine, only thyroid doses are considered in this analysis. Consideration of whole-body doses resulting from release of fission product gases and other volatile fission products leads to less severe consequences but similar general conclusions.

8.4.2 Activity Release

Paragraph 8.3.1 discusses the rupture of the main coolant system accident. As indicated, fuel melting will not occur since the core will remain covered with water and little or no damage to the fuel cladding will result. Therefore, the radioactive iodine released will be only that contained in the reactor coolant prior to the rupture. During normal operation, coolant activity levels are estimated to be a maximum of $0.4 \mu\text{c/ml}$, as shown in Table 8-1, with most of this activity resulting from crud and with fission products amounting to only $0.03 \mu\text{c/ml}$ (about 30 curies total). However, it is expected that the plant could continue to be operated for some time with activity levels considerably higher than this due to, for example, a greater number of fuel cladding defects. The upper limit of iodine activity in the coolant during operation is estimated to be $0.7 \mu\text{c/ml}$. This is about twice the estimated activity due to crud and is equivalent to release of 10 percent of the iodine produced in 10 fuel pins with continuous cleanup by the coolant purification system.

Upon rupture of the main coolant system, all of the iodine contained in the coolant which flashes to steam is assumed to be released to the brine heater pit. It is then vented with the steam through the emergency relief duct directly to the stack. Based upon the quantity of steam

estimated to be released, as indicated in paragraph 8.3.1, approximately 10 curies of iodine would be released. The remaining iodine in the coolant and any iodine released from the fuel subsequent to the steam release probably will remain in the coolant. However, even if additional iodine is released from the water to the air in the brine heater pit subsequent to the steam release, it would pass through the pit ventilation system where most of it would be removed by the ventilation filter system. It is therefore assumed that 10 curies of iodine is released over an interval of a few minutes following the accident. No decay of iodine activity is assumed before the iodine reaches a dose point.

8.4.3 Atmospheric Dispersion

Since the iodine released during the accident is vented to the stack and released to the environment from the top of the stack, atmospheric dispersion will reduce its concentration considerably before it reaches the nearest dose point. Estimates of atmospheric dilution for various atmospheric conditions are obtained using the methods and data of Gifford.⁽²⁰⁾

An effective stack height of 140 feet was assumed based on the design stack height of 200 feet minus 60 feet, the approximate height the surrounding terrain rises above plant grade. No effect of stack effluent rise due to exit velocity and temperature difference was assumed.

A wind speed of three mph was assumed. Data from various southern California stations show that winds exceed three mph about 80 percent of the time. Since the calculated dose is inversely proportional to wind speed, higher wind speeds would result in proportionately lower doses.

The calculated thyroid doses based on release of 10 curies of radioactive iodine from the stack are shown in Figure 8-1 as a function of distance downwind for a range of atmospheric conditions. It can be seen that the maximum ground level concentration occurs during the most unstable conditions, when vertical diffusion quickly brings the pollutants to ground level. As the atmospheric lapse rate and vertical mixing decrease, the maximum ground level concentration decreases and the point at which the maximum concentration occurs moves away from the stack.

This situation is different from many analyses of reactor accidents in which fission products are assumed to be released at ground level from low-leakage containment structures. In this case, stable atmospheric conditions result in the greatest calculated off-site doses. If the 10 curies of iodine released in the reference accident is released at ground level, with slightly stable atmospheric conditions the thyroid dose at the site boundary (assumed to be 500 feet) would be approximately 4.5 rem instead of the maximum of approximately 60 millirem indicated on Figure 8-1 for the stack release.

FIGURE 8-1

THYROID DOSE AS A FUNCTION
OF DISTANCE DOWNWIND FOR
VARIOUS ATMOSPHERIC CONDITIONS

8.4.4 Off-site Doses

The doses shown in Figure 8-1 were calculated based on the assumptions indicated above and using the general concentration to dose conversion methods given in TID-14844⁽²¹⁾. Both the maximum dose indicated in Figure 8-1, approximately 60 millirem, and the 4.5 rem. dose resulting from a ground release are insignificant compared with the 300 rem used as a reference thyroid dose for accident evaluations in the AEC site criteria.⁽²²⁾

Based on the above analysis, it is concluded that the environmental consequences of accidents postulated for this reactor are negligible.

Section 9

10 AND 200 MGD PLANTS

9.1 GENERAL

Plant layouts and cost estimates based on the reference 50 mgd plant conceptual design were made for plants of 10 and 200 mgd output to determine the effect of plant size on the cost of the product water. The basic concepts and most design details were the same in all plants. Changes were made to the 10 and 200 mgd plants only when required for size or to use some obvious advantage. While this approach suggests that the 10 and 200 mgd plants are probably not as well optimized for their size as the 50 mgd plant, it does permit valid comparisons to be made. It also was the most expedient approach in the time available and facilitated cost estimating of the 10 and 200 mgd cases.

This section outlines the changes which were made to the reference 50 mgd plant design to arrive at the conceptual designs of the 10 and 200 mgd plants. Differences rather than similarities between the plants are described since most of the plant description in the preceding sections applies to the 10 and 200 mgd plants as well as to the reference 50 mgd plant. All temperatures and most vertical dimensions remain the same in all three cases.

9.2 10 MGD PLANT

The site plan, reactor building arrangement, and a vertical section through the reactor and brine heater pits for the 10 mgd plant are shown in Drawing EA 8.

This plant has one-fifth the output of the reference 50 mgd plant. Because the cost of energy from this plant will be higher, it was assumed the performance ratio of the plant could be increased slightly and the required power level of the reactor was decreased by slightly more than a factor of 5 to 70 Mwt. The reactor core has an equivalent diameter of about 3.5 feet, and is approximately 4.5 feet long. The reactor tank is nine feet in diameter and is still 150 feet in height. The main coolant system consists of a single loop containing two half-capacity main coolant pumps (3500 gpm each) and three brine heaters in series, one for each stage. The reactor coolant and brine temperatures and the proportion of water produced in each section of the water plant are the same as in the reference case.

The reactor pit and brine heater pit, while still the same depth as in the reference case, have been reduced in inside diameter to 13 feet and 20

feet respectively. The spent fuel pit has a 14-foot inside diameter. The reactor building and control-administration building have been reduced in total area by about 20 percent.

To improve the plant performance ratio, the number of stages in the water plant has increased somewhat but the total water plant length remains approximately the same as in the reference case. The width of the evaporator stages has been reduced by about a factor of 5 so that the total evaporator surface area is reduced from the reference case in proportion to the plant output. Since this results in a long and narrow water plant design, the stages were laid out in a U-shaped arrangement to provide a more compact plant layout. Other arrangements could be used as well. Since the evaporator stages are now small enough to permit access to the tube bundles by lifting off the tops of the evaporator tanks with a crane, the hotter stages are not stepped downhill to provide access for the tube pulling as in the reference plant design. The total site area required for the 10 mgd plant is approximately 20 acres.

All other significant plant design features are essentially the same as in the reference plant case, scaled down as appropriate in proportion to water output or reactor power level.

9.3 200 MGD PLANT

The site plan and the reactor building plan and section for the 200 mgd plant are shown on Drawings EA 9 and EA 10, respectively.

For the 200 mgd plant, the performance ratio was assumed to be the same as for the reference case and reactor power was scaled up in proportion to water plant output, i. e., to 1600 Mwt. A three loop reactor plant design was assumed with each loop having one-third greater power output than the single loop of the reference plant design. Each loop contains the same three stages of brine heaters as in the reference case. A single large heater rather than two parallel heaters is used for the first stage to improve the equipment layout for this size plant. Each reactor plant loop feeds a separate water plant, each consisting of three sections as in the reference case but each one-third larger. Again, as in the 10 mgd plant, the reactor coolant temperatures, the brine temperatures in each stage, and the ratio of water output among the sections are the same as in the reference case. Each reactor plant loop contains only one main coolant pump of 53,000 gpm capacity. Pumps of this size are in common usage and a single pump per loop is considered to provide sufficient reliability since there is more than one loop.

The reactor core is approximately 10.5 feet in equivalent diameter and 10 feet long. The reactor tank is 16 feet in diameter and still approximately 150 feet in height. The reactor tank and the three reactor plant

loops, containing a total of nine brine heaters and three main coolant pumps, are contained in a single caisson pit with a circular cross-section and 85 feet in diameter. Pits of other cross-sectional shapes were considered, but for caissons of the size required, a circular or nearly circular shape offers the greatest volume at the lowest cost. A caisson with a rectangular cross-section would require cross-bracing walls which would increase the surface area and hence the friction during sinking and would thus increase the amount of concrete required to provide sufficient weight for sinking. The reactor tank is included within the circular cross-section and separated from the brine heaters by a concrete shield poured after the caisson is sunk. This permits the caisson to have a regular shape rather than the eccentric shape used in the reference case.

The reactor building is increased somewhat in size to accommodate the larger reactor tank and spent fuel pit. The radwaste vault area is increased substantially because the quantity of radwaste to be handled is approximately proportional to the volume of the reactor system. The control and administration building is increased slightly in area. The brine heater pit cover is in two sections because of the much greater span required and so that each section may be rolled out of the way while still permitting the reactor plant to be located as close as possible to the water plant.

Each of the three separate water plants has the same number of stages and is the same in total length as the reference case but the evaporator stages are one-third wider and provide one-third greater surface area. The three plants are arranged in parallel with the direction of flow perpendicular to the coastline. Although not indicated on the site plan, it might be possible to arrange the three sections of each independent plant so that the steel walls separating each section could be eliminated. These walls are provided in the reference case to permit plant operation at reduced capacity if one of the sections is shut down because of leakage in a brine heater. For the 200 mgd case, with three independent water plants, there is the capability of operating at reduced output without separating the sections of each plant.

A total site area of approximately 80 acres is required for the 200 mgd plant.

Section 10

ECONOMIC EVALUATION

10.1 GENERAL

Since the deep-pool reactor is a single-purpose heat source for use in water desalting, the ultimate objective of an economic evaluation is an estimate of the cost of water from the total facility. The Contract Scope of Work requires estimates of the capital cost, operating cost, and unit cost of heat of the reactor plants required to produce approximately 10, 50 and 200 mgd and the related unit costs of water. The Contract Scope of Work also states that major emphasis should be placed upon the conceptual design, feasibility determination, and cost estimates for the reactor portion of the total facility. Thus, this study was devoted primarily to the definition of the features of the complete reactor plant and those components of the water plant which differ from those of a "conventional" multistage flash evaporator plant.

The reference case is a nominal 50-mgd plant and most of the engineering detail and conceptual design definition is provided for this case. A detailed cost estimate was developed only for the reactor plant portion of the reference plant. Less detailed estimates were made for the water plant, based principally on adjustment from other recent work to account for differences in size, number of stages, and total heat transfer area of the evaporator-condenser units, and for components which are somewhat unique to this design, such as the brine heaters and the auxiliary boiler for steam jet air ejectors. The 10- and 200-mgd plants are described only in terms of changes from the reference case. The estimates for the 10- and 200-mgd plants were developed by adjustment from the reference case.

In addition to the capital cost, preliminary estimates of operation and maintenance costs and owner's costs were developed. The nuclear fuel costs were provided by COO.

It should be emphasized that these costs are of a preliminary nature and are based upon the specific assumptions outlined in this section.

10.2 CAPITAL COSTS

10.2.1 Basis

The following basic criteria were used in developing the cost estimates:

1. The reactor is the second of its kind; a previous plant of this type has been built and operated. It is assumed that operation of the first plant has shown that the design described in this report is adequate; as a consequence, no development is required for these "second generation" plants.
2. The plant is built under a turnkey arrangement with an architect-engineer (A-E) as prime contractor responsible for engineering, procurement and construction, and with a nuclear engineering firm as a subcontractor to the A-E.
3. The estimates are based on present day (early 1965) costs. No allowances are made for escalation.
4. The general reactor plant facilities are equivalent in standards to those units now being proposed for commercial operation.
5. The water plant estimate is based on present-day technology.
6. The estimates are based on the data developed and presented in this report. Where information or definition was lacking, the estimate was evaluated from other similar plants wherever possible, from scope adjustments to estimates which were prepared for recent water plant studies, and from estimates derived from recent studies of available southern California sites which are typical of the one described in this report. No quotations were solicited; however, preliminary information or opinions were obtained on the caisson price from B. Gerwick and on the circulating pumps from Byron Jackson.
7. Estimated costs of the following equipment and services were provided by COO:
 - a. Fuel.
 - b. Core structure and shielding - Direct cost delivered to the job site without installation.

Items covered in this category include canned lead gamma shield, borated steel thermal shield, upper fuel guide plate, lower grid plate, core support stand, control rod guide structure, and miscellaneous hardware associated with these items.
 - c. Control rods and drives - Direct cost delivered to the job site without installation.

Included in this category are the magnetic jacks, dashpots, position indicators, power supplies, extension pipes and latching mechanisms, control rods and miscellaneous hardware.

- d. Nuclear instrumentation - Equivalent to a construction subcontract cost including the furnishing and installation of the equipment but not the engineering.

Systems covered in this category include the main control panel assembly, rod control system relay cabinet, nuclear instrumentation rack, nuclear instrumentation equipment mounted both in the control room and in the reactor building, and the startup neutron source and its irradiation. (The cost figure provided by COO was taken from the draft of the final Elk River Reactor cost reports. Since the Elk River Reactor is a smaller reactor than the reference case in this study, and since cable lengths and other relationships appear to be somewhat more extensive and costly than those for Elk River, the cost was adjusted upward somewhat from the figure provided by COO.)

- e. Refueling equipment - Direct cost delivered to the job site without installation.

This equipment includes the refueling bridge and jib crane, closed circuit TV system including cables and reel, fuel elevator basket and drive, fuel handling tool, and miscellaneous items.

- f. Nuclear engineering.

COO provided the estimated cost of the nuclear engineering subcontract. The following items are included in this subcontract:

- Management of the entire nuclear portion of the project.
- Preliminary and detailed nuclear equipment design, drawings, and specifications for such items as: core physics, core thermal-hydraulic analysis, fuel, shielding, control rods and drives, core support structure, refueling equipment, nuclear instrumentation, radiation monitoring, and safeguards analysis.
- Providing information or functional criteria for non-nuclear components and structures.

- Review of appropriate A-E engineering.
- Preparation of operating manuals covering the nuclear portion of the plant.
- Routine operator training.
- Technical direction of reactor equipment installation, reactor startup and initial operation.

10.2.2 Scope

The estimate covers the cost for those items which might be a part of a turnkey engineer-constructor contract for the combined facility. It includes an allowance for normal construction contingencies but no allowances for changes in scope, materials, addition or duplication of equipment, etc., which are not described or implied in this report. Items of cost which are normally considered owner's costs and excluded from the capital cost estimate include the following:

1. Site selection surveys, borings, relocation, landscaping, etc.
2. Spare parts and operating supplies.
3. Ad valorem taxes.
4. Permits, licenses and fees, including the cost of obtaining them, and any engineering assistance provided.
5. Nuclear startup and conventional plant operating manuals.

Operating manuals for the nuclear portion of the plant are assumed to be provided by the nuclear engineer subcontractor. Owner's engineers are assumed to perform nuclear startup and prepare operating manuals for the remainder of the plant.

6. Temporary power substation and power consumed during construction.
7. Client engineering and construction services and client overhead.
8. Interest during construction.
9. Owner's contingency.

10.2.3 Discussion

Table 10-1 shows the capital cost breakdown for the 400 Mwt reactor for the reference 50-mgd plant. These costs cover total field costs only and exclude engineering, procurement and inspection, construction contingency, and fee. The account numbers shown correspond to those used in the AEC Code of Accounts, TID 7025⁽²³⁾.

The primary interface selected for the division between the reactor and water plant occurs at the brine heaters. The brine heaters and brine piping in the brine heater pit, but not the pit itself, are arbitrarily assigned to the water plant. With the exception of Account 213, Office, Shop, Warehouse, and Access Control, all items appear to be clearly assignable to the reactor plant. Account 213, however, includes all office, laboratory and control room space for the combined facility.

Account 211 includes sewers, paving, fencing, fire protection, lighting, etc.

Account 213 includes the control-administration building, miscellaneous foundations, painting, services, etc.

Account 219 includes the reactor and brine heater caisson (at a cost of approximately \$1,150,000), superstructure services, paint, caisson lighting, radwaste vault, and all controlled ventilation structures such as the reactor and refueling area enclosure, brine heater pit cover, reactor pit cap, etc.

Account 221 includes reactor tank internals, control rod drives, reactor tank, and reactor supporting facilities such as reactor tank sway braces, supports, etc.

Account 222 includes the main coolant pumps, piping and valves, and reactor auxiliary systems such as coolant purification, spent fuel pit cooling, decay heat removal, and safety injection. This account includes all piping, valves and insulation for the systems.

Account 223 includes equipment contained in the COO list plus fuel transfer trough and gates, auxiliary hoist, installation of fuel transfer bridge crane, TV system, etc.

Account 225 includes all the radwaste equipment, piping, pumps and valves.

Account 226 includes the instrumentation covered by the COO estimate plus all other reactor plant instrumentation and monitors, including tubing, wiring and installation.

TABLE 10-1

REACTOR PLANT CAPITAL COST BREAKDOWN
400 MWT DEEP-POOL REACTOR
FOR 50-MGD DESALTING PLANT

<u>Account Number</u>		
21	<u>Structures and Improvements</u>	
	211 Ground Improvements on-site	\$ 272,100
	213 Office, Shop, Warehouse, and Access Control	229,300
	219 Reactor and Brine Heater Containment	1,630,500
22	<u>Reactor Plant Equipment</u>	
	221 Reactor Equipment	772,200
	222 Heat Transfer Systems	973,700
	223 Nuclear Fuel Handling and Storage Equipment	115,200
	225 Radioactive Waste Treatment and Disposal	200,300
	226 Instrumentation and Control	618,500
	227 Feed Water Supply and Treatment	11,700
	229 Other Reactor Plant Equipment	65,100
24	<u>Accessory Electric Equipment</u>	818,600
25	<u>Miscellaneous Power Plant Equipment</u>	147,100
	Startup	<u>120,000</u>
	TOTAL	\$5,974,300

Account 227 assumes a simple water supply system with no separate water treatment facilities.

Account 229 includes the stack and auxiliary crane hoist.

Account 24 includes installation, connection and support of all electrical equipment for the reactor plant, including the main, reserve and station service transformers. The oil circuit breakers and main disconnects are shown under common facilities in Table 10-2.

Account 25 includes the station crane, air compressors, compressed air piping, etc. It also includes an allowance of \$50,000 for reactor plant lab equipment. An additional allowance of approximately \$150,000 is made under major common plant facilities.

Of the total shown in Table 10-1, \$669,000 of equipment and services was estimated by COO.

Table 10-2 shows the estimated capital cost for the 10-, 50- and 200-mgd plants. The basis for the 50-mgd estimate is outlined below:

Item I, Reactor Plant, is the number brought forward from Table 10-1.

Item II, Water Plant, includes under Part A the basic plant including brine heaters and product water pump station; and under B, the complete sea water intake structure and 1500 ft marine intake and discharge pipelines. In addition to the normal components of the multistage water plant, this estimate includes the cost of a package boiler and accessories for providing steam to drive the air ejectors. A special allowance has been made to cover the added complexity of the brine heaters which use double tubesheet construction.

Item III, Major Common Facilities, covers those items of cost which cannot be appropriately assigned entirely to either the reactor plant or the water plant. They were estimated to approximately the same level of detail as was the water plant. Item III A, Site Preparation, includes all excavation, sea wall, slope protection, access roads, etc. Item III B, Electrical Substation, includes both the 220 KV and the 33 KV service at the substations to the stepdown transformers. Item III C gives an allowance for a combination shop and warehouse structure complete with equipment.

Item IV, Reactor Engineering is a COO estimate and includes the complete costs of a subcontract for reactor engineering, as indicated in paragraph 10.2.1, item 7.f.

Item V covers the items shown in the Table and is estimated in accordance with the basis and scope as previously defined.

TABLE 10-2
CAPITAL COST ESTIMATE

<u>Plant Size</u>	<u>10-mgd</u>	<u>50-mgd</u>	<u>200-mgd</u>
	Costs \$10 ³		
I. Reactor Plant	\$ 3,989	\$ 5,974	\$ 13,596
II. Water Plant			
A. Basic Plant including Brine Heaters	7,700	29,200	112,400
B. Intake Structure and Marine Pipelines	1,900	3,700	12,500
III. Major Common Facilities			
A. Site Preparation	1,100	1,600	4,800
B. Substation	300	300	300
C. Shop and Warehouse	150	200	300
IV. Reactor Engineering Including Fee	800	900	1,100
V. A-E Engineering, Procure- ment, Inspection, Construc- tion Contingency and Fee	<u>3,350</u>	<u>6,000</u>	<u>15,500</u>
SUBTOTAL	\$19,289	\$47,874	\$160,546
Land and Land Rights	400	800	1,600
Owner's Costs	<u>2,350</u>	<u>4,850</u>	<u>16,200</u>
TOTAL COSTS	\$22,039	\$53,524	\$178,346

The subtotal of these five items gives the total estimated contract price for the facility. In addition, an allowance for land and for owner's costs was made to estimate the total project cost. It is quite difficult to estimate the land and land rights cost for this plant. Estimated unit costs of land (cost per acre) as well as estimated land area requirements for this plant are quite uncertain. Assuming that 40 acres of land is required and that the land has an average value of \$20,000 per acre, the estimated land cost would be \$800,000. However, it would not be unrealistic to assume values 5 to 10 times higher for coastal property in southern California. Owner's costs including interest during construction were assumed to be approximately 10 percent of the total capital costs, based on an estimated field construction period of approximately 24 to 28 months.

The 10- and 200-mgd estimates were obtained by adjusting the 50-mgd estimate. In each case the items are identical in scope to those for the 50-mgd plant. The cost of land and land rights is adjusted based upon estimates of the changes in property requirements for the two plants using the same unit land costs that were used for the 50-mgd plant (\$20,000 per acre). Owner's costs are arbitrarily increased to 12 percent for the 10-mgd plant to allow a higher percentage for owner's engineering and other owner's costs which do not scale directly with costs of the project. However, for the 200-mgd plant owner's costs are maintained at 10 percent since the longer construction period assumed to be required for this plant will increase the interest during construction enough to offset the reduced percentage for owner's engineering costs.

10.3 FUEL COSTS

10.3.1 General

The fuel cost data for this study were supplied by COO. A discussion of the assumptions used and of the results are presented in the following sections.

10.3.2 Assumptions used in Fuel Cost Estimate

The calculation of costs for the 400-Mwt reactor was based on the following:

- Reactor
 1. Reactor Power - 400 Mwt
 2. Fuel Loading - 25,000 KgU
 3. Discharge Burnup - 20,000 mwd/t
 4. Plant Factor - 90 percent

5. Annual Refueling
 6. Initial Enrichment - 2.75 percent
 7. Discharge Enrichment - 1.15 percent
 8. Conversion Ratio - 0.55
- Cost
 1. Fabrication and Fresh Shipping - \$100/KgU
 2. Uranium Inventory Change - 5 percent
 3. Reprocessing - NFS Contract for Small Lots
 4. Pu Credit - \$10/gm
 5. Working Capital - 5 percent (24)
 6. Spent Fuel Shipping - West Coast Reprocessing Plant
 7. U_3O_8 - \$8/lb
 8. Separative Work - \$30/KgU
 9. Cascade Tails Assay - 0.253 percent U-235

10.3.3 Results

The calculated fuel cost for the 400-Mwt reactor using the above assumptions is $17\text{¢}/10^6$ Btu. This is representative of presently quoted fuel costs, recognizing no attempt was made to optimize the core for the low-temperature and pressure conditions pertinent to this reactor.

Fuel costs for the 70-Mwt and 1600-Mwt reactors were obtained by adjusting the $17\text{¢}/10^6$ Btu value to reflect the higher unit costs associated with the smaller reactor, and the relatively small improvement expected in the larger reactor.

Representative present day and expected range of future fuel costs are tabulated below. The present day values were used in determining the cost of water for the three plants.

<u>Reactor Size</u> <u>Mwt</u>	<u>Fuel Cost (¢/10⁶ Btu)</u>	
	<u>Present</u>	<u>Future</u>
70 (10-mgd)	25	16
400 (50-mgd)	17	11
1600 (200-mgd)	16	10

The future costs are representative of what can be achieved in the later years of plant operation as a result of improved core designs and fuel management, and a greater volume of fuel fabrication and reprocessing in the industry (references 25, 26, 27).

10.4 ANNUAL COSTS OF OPERATION AND MAINTENANCE

10.4.1 General

In order to arrive at the costs of energy and product water it is necessary to estimate and allocate the annual costs of operation and maintenance. The following discussion outlines the procedure used to develop these costs for the 50-mgd plant. Similar procedures were employed to obtain comparable figures for the 10- and 200-mgd plants.

10.4.2 Labor

The operating labor estimated to be required for the plant is tabulated below:

<u>Water Plant</u>		<u>Reactor Plant</u>	
Shift Personnel	9	Shift Personnel	14
Chemist	1	Nuclear Engineer, Reactor	
Mechanical Engineer	<u>1</u>	Operation Supervisor	1
		Radiochemist, Health Physicist	1
Total	11	Mechanical Engineer	1
		Electrical Engineer	1
<u>Common</u>		Instrumentation Engineer	<u>1</u>
Plant Superintendent	1	Total	19
Assistant Superintendent			
and Water Plant Operating			
Supervisor	1		
Clerical	1		
Custodial	<u>1</u>		
Total	4		

In allocating costs between the reactor plant and the water plant, the operating personnel common to both facilities were divided equally. Therefore, it was estimated that the reactor plant operations requires the equivalent of 21 people and the water plant requires an equivalent of 13 people. A labor rate slightly over \$10,000 per man-year was used to estimate labor costs.

10.4.3 Maintenance and Other Costs

Reactor plant maintenance is assumed to be performed partly by maintenance personnel and partly by subcontract. Other operating cost components for the nuclear plant include power, estimated at 6 mills/kwh, ion exchange resins at \$30,000/yr, and working capital estimated at approximately 2.5 percent of the average annual operating expense for the nuclear plant. The nuclear property and liability insurance for the 50-mgd plant is estimated at \$300,000 per year based on 0.7 percent of the reactor plant cost and 0.4 percent of the average fuel inventory for nuclear property insurance, a \$200,000 allowance for NELIA/MAELU insurance and \$30.00/Mwt/yr for Federal Government liability insurance. For the 10- and 200-mgd plants the allowance for NELIA/MAELU was prorated.

For the water plant, the additional operating cost items include power and sulfuric acid. Sulfuric acid is estimated at 2.8¢/1000 gallons of product water. Working capital for the water plant was estimated on a basis similar to that for the nuclear plant.

All O & M estimates are based on the assumption that the plants operate at a plant capacity factor of 90 percent. Results of these estimates are shown in Table 10-3.

10.5 COSTS OF ENERGY AND WATER

In order to estimate the cost of energy from the various reactors, it is necessary to arbitrarily allocate a portion of the common facilities costs and engineering indirects to the reactor plant. For this study, it was assumed that total reactor plant costs include: total field costs for the reactor plant, land, reactor engineering, a prorated portion of the A-E costs (approximately 1/3 for the 50-mgd plant, somewhat more for the 10-mgd plant and less for the 200-mgd plant), and owner's costs allocated in proportion to the capital cost (paragraph 10.3). The capital charge rate used was seven percent. Energy costs are based upon 90 percent plant capacity factor. The estimated cost of energy for the 10-mgd, 50-mgd, and 200-mgd plants is shown in Table 10-4.

The estimated cost of water was obtained using the energy costs as discussed above, a 7 percent capital change rate for the water plant and

the O & M costs shown in Table 10-3. Table 10-4 summarizes these costs for the 10-mgd, 50-mgd, and 200-mgd plants. Capital costs and product water costs are plotted as a function of plant output in Figures 10-1 and 10-2.

TABLE 10-3
ANNUAL OPERATING AND MAINTENANCE COSTS

<u>Plant Size</u>	<u>10-mgd</u>	<u>50-mgd</u>	<u>200-mgd</u>
NUCLEAR PLANT	Cost \$10 ³ /yr		
Operating Labor	\$175	\$ 220	\$ 245
Maintenance Labor and Material	70	100	165
Electrical Power	50	190	670
Ion Exchanger Resin	10	30	120
Nuclear Property and Liability Insurance	180	300	520
Working Capital	<u>25</u>	<u>60</u>	<u>220</u>
TOTAL	\$510	\$ 900	\$1,940
WATER PLANT			
Operating Labor	\$115	\$ 130	\$ 145
Maintenance Labor and Material	55	125	310
Electric Power	200	945	3,750
Sulfuric Acid	90	460	1,840
Working Capital	<u>15</u>	<u>40</u>	<u>155</u>
TOTAL	\$475	\$1,700	\$6,200

TABLE 10-4
ENERGY AND WATER COSTS

<u>Plant Size</u>	<u>10-mgd</u>	<u>50-mgd</u>	<u>200-mgd</u>
ANNUAL ENERGY COSTS, \$10 ³ /yr			
Reactor Plant Capital	520	745	1, 510
Fuel	470	1, 820	6, 860
Nuclear Plant O & M	<u>510</u>	<u>900</u>	<u>1, 940</u>
TOTAL	1, 500	3, 465	10, 310
UNIT ENERGY COST, \$/10 ⁶ Btu			
	0.80	0.32	0.24
ANNUAL WATER COSTS, \$10 ³ /yr			
Water Plant, including Common Facilities	1, 020	3, 005	10, 970
Water Plant O & M	475	1, 700	6, 200
Energy	<u>1, 500</u>	<u>3, 465</u>	<u>10, 310</u>
TOTAL	2, 995	8, 170	27, 480
UNIT WATER COST, \$/1000 gal			
	0.91	0.50	0.42

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DRAWINGS

BECHTEL CORPORATION SAN FRANCISCO		
U.S.A.E.C. CHICAGO OPERATIONS OFFICE		
CROSS SECTION		
DEEP POOL REACTOR FOR WATER DESALTING		
5395	B-A-1A

P L A N

NOTE:
DRAWINGS REDUCED FOR PUBLICATION USE GRAPHIC SCALE.

Scale 1" = 50'		Drawn A. H. H. & S.	
BECHTEL CORPORATION SAN FRANCISCO			
U. S. A. E. C. CHICAGO OPERATIONS OFFICE			
50 MGD PLANT SITE PLAN			
DEEP POOL REACTOR FOR WATER DESALTING
JOB No.	DRAWING No.	REV.
	5395	E-A-1

SECT. A-A

SECT. J-J

SECT. K-K

GRAPHIC SCALE 1/8" = 1'-0"

NOTE: DRAWINGS REDUCED FOR PUBLICATION USE GRAPHIC SCALE.

BECHTEL CORPORATION SAN FRANCISCO		
U.S.A.E.C. CHICAGO OPERATIONS OFFICE		
S.M.G.D. PLANT VERTICAL SECTIONS DEEP POOL REACTOR FOR WATER DESALTING		
5395	EA-2

SECT. D-D
1/8" = 1'-0"

SECT. H-H
1/8" = 1'-0"

SECT. C-C
1/8" = 1'-0"

SECT. F-F
1/8" = 1'-0"

SECT. F-F
1/8" = 1'-0"

NORTH ELEVATION
1/16" = 1'-0"

EAST ELEVATION
1/16" = 1'-0"

SOUTH ELEVATION
1/16" = 1'-0"

WEST ELEVATION
1/16" = 1'-0"

NOTE: CUT ON SHT. NO. 215 SHOWN ON SHT. NO. 215

NOTE: DRAWINGS REDUCED FOR PUBLICATION USE GRAPHIC SCALE.

NOTED	NOTED
BECHTEL CORPORATION SAN FRANCISCO	
U. S. A. E. C. CHICAGO OPERATIONS OFFICE	
50M.G.D. PLANT HORIZONTAL SECTIONS AND ELEVATIONS	
DEEP POOL REACTOR FOR WATER DESALTING	
5395	E-A-3

ENLARGED TROUGH DETAIL
1/2" = 1'-0"

SECT. A
1/2" = 1'-0"

SECT. B
1/2" = 1'-0"

PLAN
EL. (+49'-0")
1/8" = 1'-0"

SECT. C
1/8" = 1'-0"

TOP VIEW
1/2" = 1'-0"

SECT. CUT
CUT ON SHT. NO. 515 SHOWN ON SHT. NO.

NOTE:
DRAWINGS REDUCED FOR PUBLICATION USE GRAPHIC SCALE.

NOTED		REVISIONS	
BECHTEL CORPORATION SAN FRANCISCO			
U.S.A.E.C. CHICAGO OPERATIONS OFFICE			
REFUELING ARRANGEMENT DETAILS			
DEEP POOL REACTOR FOR WATER DESALTING			
5395	E-A-5	DATE	BY

SCALE NONE	DESIGNED	DRAWN J. Hine
BECHTEL CORPORATION SAN FRANCISCO		
U.S.A.E.C. CHICAGO OPERATIONS OFFICE		
50 M.G.D. PLANT ELECTRICAL SINGLE LINE DIAGRAM DEEP POOL REACTOR FOR WATER DESALTING		
JOB No.	DRAWING No.	REV.
5395	DA7

FIRST FLOOR PLAN
1/8" = 1'-0"

SECTION A
1/8" = 1'-0"

SITE PLAN
1" = 50'

CUT ON SHT. NO. 213
SECTION SHT. NO.

NOTE: DRAWINGS REDUCED FOR PUBLICATION USE GRAPHIC SCALE.

BECHTEL CORPORATION SAN FRANCISCO	
U.S.A.E.C. CHICAGO OPERATIONS OFFICE	
10 M.G.D. PLANT BLDG EQUIPMENT ARRANGEMENT, SITE PLAN, SECTION	
DEEP POOL REACTOR FOR WATER DESALTING	
5395	E-A8

NOTE: DRAWINGS REDUCED FOR PUBLICATION USE GRAPHIC SCALE.

BECHTEL CORPORATION SAN FRANCISCO		
U.S.A.E.C. CHICAGO OPERATIONS OFFICE		
200 M.G.D. PLANT SITE PLAN		
DEEP POOL REACTOR FOR WATER DESALTING		
SHEET NO. 5395	DRAWING NO. E-A9	REV.