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AEC RESEARCH AND DEVELOPMENT REPORT

PRELIMINARY SAFETY EVALUATION OF THE SODIUM REACTOR EXPERIMENT

Preliminary

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I. INTRODUCTION

Reactors of many types have been suggested as heat sources for the central-station-type of steam power plant. One of the most promising for early development is the sodium-cooled, graphite-moderated, thermal reactor. In a project sponsored jointly by the Atomic Energy Commission and North American Aviation, Inc. it is proposed to conduct a "Sodium Reactor Experiment" ("SRE") at the North American Field Test Station at Santa Susana, California. This experiment will involve the construction and operation of a 20 MW, sodium-cooled, graphite-moderated reactor. The purpose of this experiment is to:

1. Demonstrate the capability of the Sodium Graphite Reactor system.
2. Evaluate and extend the performance limits of low-enrichment U fuel elements.
3. Evaluate and extend the capabilities of the Th-U^{235} (Th-U) alloy as a reactor fuel.

The proposed site was presented before the Reactor Safeguards Committee at its 19th meeting on December 16, 1952 at Arco, Idaho. At this time it was under consideration as the possible site of a Plutonium and Power Reactor Experiment ("PPRE"), which was a 30 MW pilot reactor plant for the simultaneous production of plutonium and power. The basic site data which were presented in NAA-SR-Memo-414, Part 2 have, of course, in no way changed since this earlier presentation, however, some additional meterological data has been obtained.

II. CONCLUSIONS

A study has been made of the possible hazards associated with the operation of the proposed reactor at the proposed site. We have considered the consequences of equipment failures and malfunctions, the events which might follow operating errors by personnal, Delete [redacted] and the incidents which might arise from natural causes. In particular we have studied:

(1) Reactor nuclear runaways (uncontrolled release of excessive amounts of nuclear energy)

(2) Interruption to the flow of coolant in the reactor (failure of the normal means of heat removal)

(3) Sodium fires

(4) Other chemical reactions

(5) Release of stored energy

Delete [redacted]

(7) The effects of earthquakes

It is concluded that the sodium reactor experiment presents no serious hazard to the general public and to the surrounding area under the most severe conditions of accident Delete [redacted] which may be realistically assumed to be possible.

The sodium-graphite-low enrichment uranium system has an inherent negative temperature coefficient of reactivity. In addition, the reactor, the reactor controls, and the cooling system have been so designed as to prevent accidents insofar as this is possible and practical and to limit the consequences of any accidents. Furthermore the system has been so arranged that the possible but highly improbable incident in which some fuel does melt will present no serious hazard to the general public.

There has also been calculated the "ultimate hazard" — the purely hypothetical and not at all realistic case — wherein all the radioactivity in the reactor is released to the atmosphere. Even in this case it is shown that the situation is not catastrophic, and that there is still a "factor of safety", over conditions assumed acceptable on an emergency basis.

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III. DESCRIPTION OF SRE SITE

A. Location

The proposed site is located 30 miles northwest of downtown Los Angeles, 6 miles west of Chatsworth and 3 miles south of Santa Susana, California, in the Simi Hills.

The elevation of the reactor site is approximately 1850 feet, and the maximum elevations of the Simi Hills are about 2400 feet. The elevation of the San Fernando and Simi Valley floors at the base of the hills is approximately 900 feet.

The Simi Hills are a very rugged outcropping of sandstone strata. The hilly surrounding barren area which provides the required 1.4 mile radial clearance for a 20 megawatt reactor. The proposed site is unique in that it provides the isolation and security required for a reactor facility while being in the proximity of scientific resources (See Fig. 1).

B. Climatology

The Los Angeles basin is in a semi-arid region controlled for the most part by the semi-permanent Pacific high pressure cell. During the summer season the high is displaced to the north resulting in mostly clear skies with little precipitation. This, in conjunction with the diurnal thermal lows that form over the inland valleys and deserts, results in stratus during the early morning hours. Winds are mainly diurnal sea to land during the day and land to sea during the night. In the winter there is moderate precipitation and the winds are predominately from a northerly direction.

The surface winds in the Los Angeles vicinity have a diurnal and a seasonal variation due to the influence of the ocean. In addition they are greatly influenced by topography. In the case of the site, the surface wind during the cooler hours of the day will be southeasterly, corresponding to the downward slope of the terrain toward Simi Valley. The afternoon winds should tend to reflect the main ocean-continent effect, resulting in westerly winds in summer and northerly winds in winter.

The wind speeds in summer, and in winter in the absence of storms, are light. Los Angeles City Office has an annual average of only 6 miles per hour. The winds at the site should also be expected to be light, the increase in elevation not being sufficient to make material difference in winter, while in summer the winds at moderate elevations may actually be stronger than those near sea level, where the sea breeze is influential.

A subsidence inversion is present almost every day during the summer months and frequently in other months. In addition, a ground inversion is produced by radiational cooling during clear nights in fall, winter, and spring. The nocturnal inversions produced on clear nights are associated with a tendency for the cooled air to drain downward from the slopes on which the site is located. During the day, even when the subsidence inversion over lower terrain is lower than the site, the diurnal heating tends to establish a stirred layer with adiabatic lapse rate immediately above the hills.

The occurrence of fog at the ground and other limitations to visibility is strongly a function of elevation and location in this vicinity. The proposed site, being near the ridge of hills, will not be subject to local radiative fog at all, and in general will enjoy excellent visibility both summer and winter. Only when the inversion base is higher than the site will there be a chance of fog or stratus cloud. Because of cold air drainage, there will seldom be radiational fog in winter.

C. Vertical Atmospheric Cross Section

The inversion structure above the site is of prime importance. There are four patterns pertinent to diffusion of contaminants.

1. No inversion exists. The lapse rate is then unstable, allowing rapid vertical convection currents and a thorough mixing of the contaminants into a deep layer of air. It is only during the winter months that these conditions prevail for any significant per cent of the time.
2. Inversion below the site. This may persist for days. Such inversions are caused by subsiding of the air mass in the high cell, hence cannot easily be broken by daytime insolation. Percentage wise this is the most common case for the site. Under such conditions contaminants released at the site would diffuse laterally only; the inversion prevents any downward transport.

3. Inversion at or above the site. This is a rather rare occurrence. When the inversion is at or slightly above the site, then the most unfavorable situation for diffusion exists, since contaminants are prevented from upward diffusion and transport downward is very likely. However, when the inversion height is more than 4000 feet (mean sea level) then there is sufficient lateral and vertical diffusion to minimize the probability of concentrations from reaching the valley floors.
4. Nocturnal ground inversion. Such inversions form during the night on calm clear nights and rapidly dissipate soon after sunrise. During such inversions, there is cold air drainage into low-lying areas and valleys. However, this air does not reach the valley floor but rides over the valley air. Therefore, contaminants are kept aloft and do not reach the surface in appreciable amounts.

D. Wind Pattern Studies

Two types of studies have been carried out to obtain a more detailed wind flow structure in the immediate area of the site: 1) balloon tracking, and 2) aerosol diffusion.

The balloon studies confirmed the diurnal nature of the local winds during the summer. Also, the effect of topography was very evident in the devious paths followed by the balloons in the lower levels.

The aerosol diffusion studies gave a more quantitative answer to contaminant transport. This one study pictured the effect of both inversions and winds. Fortunately the period of the runs was in the early spring so that both summer and winter conditions were encountered. On the other hand a total of only 405 samples were taken, and therefore the results have only a qualitative significance. The results may be summarized as follows:

1. Very low counts were found when no inversion was present.
2. Very low counts were found when the inversion was below the site.
3. The highest counts were found when the inversions were at or near the elevation of the site.
4. Radiation inversions did not cause any large counts in the valleys due to the slow drainage rate and attrition by entrapment in gullies and ravines.

5. On days of strong turbulence counts were obtained in diametrically opposite directions, attesting to the value of turbulence in producing a broad diffusion pattern.

The highest count of 15 particles/ft^3, unfortunately occurred on a day when no radiosonde observation data were taken. However, interpolating between the preceding day and the succeeding day it is safe to say that the inversion was at or slightly above the site.

Taking all the various conditions into consideration, about 93 per cent of the time there is a lower danger of contamination of populated areas. The 7 per cent figure for the time for unfavorable situations is a liberal one including various degrees of intensity of possible contamination.

E. Hydrology

The proposed site is located on relatively flat terrain which straddles the drainage divide between the San Fernando and Simi Valleys. The runoff from the area is collected in gulleys or washes which are generally dry most of the year. The southeastern half of the proposed site drains into the Los Angeles River which empties into the Pacific Ocean at Long Beach. To the north and west, drainage follows the Simi Valley to the west and discharges into the Pacific Ocean approximately 1 mile southeast of Port Hueneme.

A portion of the local runoff which originates on the east side of the Simi Hills is captured by the Chatsworth Reservoir. This drainage area is over 2 miles from the proposed site and on the opposite side of the hills; the reservoir itself is approximately 4 miles east of the proposed site. It is a seasonal regulating and storage reservoir with a capacity of approximately 10,000 acre feet. The reservoir water is used for drinking and general domestic purposes except in the fall of the year when the water is used for irrigation. During this time potable water is rerouted from other reservoirs. This alternate supply route could be used for an indefinite period of time in case of an emergency.

Since the proposed site area is high, the immediate drainage area is relatively small, approximately 200 acres. The maximum precipitation in 24 hours with a probable frequency of occurrence once in 50 years is 8 to 9 inches. Assuming 5 inches of runoff, about 85 acre-feet of water might be expected from a single storm. During a normal year only 50 acre-feet would run off.

Since the total expected runoff is relatively small, it is proposed to make provisions for capturing all of it in a reservoir. All the surface drainage in a normal year would evaporate from the reservoir surface. The proposed reservoir would hold 200 acre-feet, although it would appear that a 100 acre-feet storage reservoir would be adequate.

The Simi Hills are composed primarily of marine sediments consisting of tightly cemented sandstones and shales (Chico formation) of Upper Cretaceous age. These are relatively impervious layers tilted such that they have a strike of approximately 325° (35° west of north) and a dip of approximately 22°.

The relatively small flats found in the area consist of alluvium of recent geological age. Generally the alluvium increases in thickness as the slope of the hillside decreases, and is a maximum thickness in the relatively flat areas and valleys. These recent alluvial fills are relatively pervious compared to the relatively impervious sedimentary strata from which they are derived.

The exposed strata have a maximum thickness of about 30 feet. In many places they are badly fractured and large monoliths have broken loose and fallen from their original strata and are scattered on the hillsides.

The fissures and fractures of the faults contain a water bearing aquifer, and wells drilled into this pervious material have been found to be very good. The water in these fissures and fractures is tightly contained by the relatively impervious rock which encloses them. The aquifers are recharged during the rainy season by rain percolating directly through the soil to the aquifer and also by the stream beds crossing the pervious fault zone which traps some of the runoff. It is possible that some of the strata could be semi-pervious and therefore allow a fluid to move parallel to the bedding plane.

Assuming that the dip of the bedding planes is consistently 22°, it is seen that the particular planes intersecting the site area would project far below the Simi Valley. These strata are sloping away from the San Fernando Valley. Therefore no drainage parallel to the bedding planes could reach a populated area. It is concluded that any radioactive material accidently released into the surrounding substrata would slowly permeate far below the land surface and remain trapped there.

Since the terrain is such that complete surface control is easily accomplished, the site appears to be well located. The sub-surface hydrology of the area also seems favorable.

F. Seismological Characteristics

The Southern California region has had a rather active history of earthquakes. Building designs and methods of construction have been developed to withstand earthquake forces. While a severe earthquake always leaves in its wake many collapsed buildings, experience has shown that other buildings in the area which were designed properly have been capable of withstanding the strongest recorded earthquake shocks.

Since the proposed reactor is to be built on solid ground several miles away from an active fault, the structure, when properly designed, will be capable of withstanding the strongest recorded earthquake shocks. As an added safety factor, mechanisms that operate when a predetermined shock intensity level is reached will be utilized to scram the reactor.

G. Population Distribution

No people reside within the exclusion radius of 1.4 miles. One and one-half miles from the reactor site is the NAA propulsion laboratory employing approximately 300 people. Within a 5-mile radius reside 3000 people, and the population within a 10-mile radius is 56,000. Most of the population within the 10-mile radius is in the San Fernando Valley, east and on the opposite site of the Simi Hills from the proposed site. Chatsworth, 6 miles east, has a population of 2,500. To the north is the Simi Valley. It includes a number of unincorporated sparsely populated areas. The total 1952 population in the Simi Valley proper is estimated at 3,280 (See Fig. 2).

The area seems to be growing at a normal rate. No sharp increase in this rate is expected because of the water shortage in the area. If it is ever found feasible to bring water into the area from some outside source to replenish the rapidly declining underground water supply, a more rapid growth could be expected to take place. However, the mountains around the valley make this task a very difficult and costly one.

IV. DESCRIPTION OF THE SRE FACILITY

A. Reactor

The reactor of the SRE facility (Fig. 3) is a 20 megawatt (nominal) unit composed of a graphite moderator and reflector, sodium coolant, and low enrichment uranium metal fuel. Essentially all the fissions are produced by neutrons of thermal energy which in the SRE is near 0.075 ev. The neutron life time is approximately 0.5 millisecond.

Each fuel element consists of a cluster of seven 6-foot long rods supported vertically from the top shield by a hanger rod and shield plug. Each fuel rod is composed of six slugs of uranium metal 3/4 inch diameter by 12 inches long. The slugs are contained within a 0.010 inch wall stainless steel tube. There is a 0.010 inch thick stagnant sodium bonding layer between the uranium and the steel cladding. The uranium will need to contain approximately 2.0 atom per cent U^{235}. The fuel elements, nominally 31 in number, are arranged on a triangular lattice with 11 inches between centers.

The graphite for the moderator and reflector is contained within thin-walled zirconium cans. The moderator cans are hexagonal prisms, approximately 11 inches across the flats and 10 feet long, containing a centrally located fuel channel 2.805 inches inside diameter running the full length of the can. The purpose of the zirconium can is to isolate the graphite from the sodium and to localize any leak that might occur. The reflector elements are similar except they contain no fuel channels.

There are 37 fuel channels; 31 will be loaded for normal operation. Out of 72 reflector cans 18 in the first row outside the core contain centrally located channels for additional flexibility and experimental purposes.

The moderator and reflector elements are contained within a steel core tank 11 feet ID by 21 feet high. The graphite stack is roughly a cylinder 11 feet OD by 10 feet high. Sodium coolant enters at the bottom of the tank and flows upward through the process channels and between the zirconium cans to a pool or reservoir above the cans. This reservoir of sodium is approximately 6 feet deep. The exit coolant temperature from each channel is monitored continuously.

The entire reactor and primary cooling system are constructed below ground level; the only access to the reactor and primary cooling system is thru the top shield. This arrangement simplifies the reactor building and the shielding and contributes to the over-all safety of the installation. The top shield is a 5-foot thick steel and dense concrete assembly containing the movable plugs from which the individual fuel elements are suspended. The reactor foundation and shield structure (side and bottom) will be constructed by pressure grouting a preplaced aggregate. This will also pressure grout adjacent soil strata making them impervious to seepage. The inner faces of the foundation-shield will be lined with a gas-tight steel membrane. An inert gas atmosphere will be maintained within it at all times.

There are to be four control-shim elements and four safety elements. Each set of four elements will control about 8 per cent Δk or roughly $10. These rods are mounted vertically and operate in thimbles located within channels similar to fuel channels but spaced between lattice points. The thimble is to exclude sodium and sodium vapor from the rod operating mechanism. The rods are actuated by lead screws driven by motors located on top of the upper neutron shield. The nuclear poison material will be 3 per cent boron steel. The safety elements are ball-type. Balls containing a nuclear poison are stored in a hopper above the core during reactor operation. When a holding magnet is de-energized, as on scram, a gate opens under the action of gravity, allowing the balls to fall into a thimble in the reactor core. An air stream is used to return the balls to the storage hopper.

Safety circuits will be provided which will cause the reactor to "Setback" or "Scram" on the following signals:

1. Short period of reactor neutron flux
2. High Reactor Flux Level
3. Loss of flow in any of the four sodium loops
4. Loss of air-flow across air blast heat exchangers
5. Low sodium level in reactor tank or surge tanks in the two secondary loops.
6. Low pressure of blanketing gas in the reactor tank or in the surge tanks or two secondary loops
7. High temperature in hot leg or cold leg of reactor
8. High fuel temperature
9. Failure of electrical power
10. Earthquake shock

The following interlocks will be incorporated in the control circuits:

1. Control rods may not be withdrawn until the poison balls of the safety elements have been removed from the thimbles and returned to the storage hoppers (safety elements cocked). On any "Scram" control rods will be driven into the "In" position automatically.

TABLE I

REACTOR DATA

Nominal power rating	20,000 kw
Fuel, Enriched Uranium metal (for 31 channel loading)	2120 kg (2.0 atom per cent U^{235})
U^{235} content for criticality	42.4 kg
Average specific power	9 kw/kg-U (or 470 kw/kg U^{235})
Initial conversion ratio (U^{235} into Pu^{239})	0.50
Maximum gross fission product beta-activity	95.5 megacuries
Maximum gross fission product gamma-activity	3×10^{18} Mev/sec
Maximum plutonium build-up	2.2 kg (220 curies)
Number of fuel elements (nominal)	31
Active fuel rod length	6.0 ft.
Effective core diameter	5.4 ft.
Lattice spacing (triangular array)	11.0 inches
Average thermal flux in fuel	2×10^{13} n/cm^2-sec
Peak thermal flux in moderator	5.5×10^{13} n/cm^2-sec (approx)
Peak fast flux in moderator	1.2×10^{14} n/cm^2-sec (approx)
Neutron generation time	0.5 millisecond
Multiplication factor, k_∞	1.260
Material buckling, $\mathcal{B}$	450×10^{-6} cm^{-2}
Resonance escape probability, p	0.87
Thermal utilization, f	0.89
Fast effect, e	1.034
Slowing down area, L_s^2	349 cm^2
Thermal diffusion area, L^2	172 cm^2

2. Coolant in the four sodium loops must be circulating at or above 50 per cent of designed full power flow before the safety elements may be cocked.
3. A specified minimum counting rate must be provided by the low level counting channel before safety rods may be withdrawn. This will assure that an extraneous source of neutrons is present, and that the counting channel is operating.

Neutron fluxes in the reactor and coolant temperatures and pressures in the reactor and cooling system will be monitored, and an audible alarm sounded when a specified range is exceeded. This will include such signals as temperature of individual fuel elements, pressure of inert gas blanket, flow in auxiliary coolant circuits, and temperatures at points along the reactor tank wall. The general philosophy is to inform the operator of every condition which represents abnormal behavior in any part of his system, yet to hold the number of scrams to the necessary minimum.

B. Cooling System

The coolant in the SRE is sodium. The SRE cooling system consists of two parallel circuits; a main circuit and an emergency circuit. Each circuit is made up of a primary and a secondary loop coupled thru an intermediate exchanger. The primary loops (main and ~~emergency~~ auxiliary) are common at the reactor only. The secondary loops are completely independent and have separate sodium-to-air heat exchangers thru which they reject the reactor heat to the atmosphere.

The main cooling system is capable of removing 20,000 kilowatts; the auxiliary ~~emergency~~ system 1000 kilowatts. The speed of the main pumps may be varied so as to maintain a constant temperature difference across the reactor for all power levels between 2 and 20 megawatts. Either of the two parallel circuits may operate if the other should fail. If the reactor tank or inner container should rupture, the reservoir of sodium is such that the sodium level will not fall below the suction lines in the reactor vessel. Thermal convection in the main circuit will permit dissipation of at least 200 kilowatts of heat. There will be, of course, an emergency power supply for the pumps and instruments.

TABLE II

COOLING SYSTEM DATA

Reactor	
Nominal reactor thermal power (31 channels)	20,000 kw
Inlet temperature	500° F
Outlet temperature (mixed mean)	960° F
Coolant flow rate	485,000 lbs/hr
Maximum coolant velocity	6 ft/sec
Maximum uranium temperature	1200° F
Pressure drop thru coolant tube	2.5 psi
Main Cooling Circuit	
Nominal thermal rating	20,000 kw
Auxiliary ~~Emergency~~ Cooling Circuit	
Nominal thermal rating	1,000 kw

C. Shield ~~Auxiliary~~ Cooling System

An Shield ~~auxiliary~~ heat removal system using the organic fluid toluene is provided to cool the biological shield lining the reactor cavity, the reactor top shield plug, and the inert gas atmosphere in the underground galleries housing the primary sodium system. Toluene has the advantages of a relatively high boiling point, compatibility with sodium, and a low thermal and radiation decomposition rate in this application. The toluene cooling system will have a capacity of approximately 100 kw or 1/2 per cent of the thermal rating of the reactor.

D. Building

The building housing the SRE facilities will have a high-bay area over the reactor and primary cooling system which is approximately 100 feet long by 50 feet wide by 45 feet high. A connecting building will provide cover for laboratory facilities and a storage pond for irradiated fuel elements. The building will be so designed that the level of the water in the storage pond will at all times be at least 4 feet below the operating face of the reactor.

The building will be constructed of steel and concrete. Wall panels will be lightweight precast reinforced concrete fastened to steel columns spaced at 16 feet centers. Individual panels will be about 8 feet high and 6 inches thick with an insulating cavity. The roof deck will be lightweight precast reinforced concrete attached to steel purlins spaced on about 8 feet centers. An insulating core is also contemplated in the roof panels.

V. EXPERIMENTAL PROGRAM

The reactor facility described was originally conceived as a tool for a long range, experimental program. The object of this program was the development of a low-cost reactor system for the generation of competitive electric power. This program was envisioned as including the study of fuels, moderators, and coolants in various configurations as well as the proof testing of auxiliary equipment.

The Sodium Reactor Experiment as presented here constitutes the first phase only of this program. The immediate objectives of the Experiment were listed above. They are:

1. Demonstrate the capability of the Sodium Graphite Reactor system
2. Evaluate and extend the performance limits of low enrichment U fuel elements
3. Evaluate and extend the capabilities of the Th-U^{235} (Th-U) alloy as a reactor fuel

In the conduct of the Sodium Reactor Experiment no basic changes will be made in the reactor configuration or in the major components. In general, the successive, individual experiments will involve only the change of one or more fuel elements or the substitution of test specimens for fuel elements in selected fuel channels. The scope of the proposed experiment is best defined with respect to the areas of investigation:

1. Coolant

   The only coolant to be studied will be sodium. Experiments will be made to determine the upper limitation on temperature, that is, temperatures will be raised above the design point.

2. Moderator

   The experiment will utilize only zirconium canned graphite for the reactor moderator; study of other moderators will be limited to irradiation of specimens in capsules inserted in the fuel channels.

3. Fuels

   The principal part of the experimental program will relate to fuels. Experiments will investigate various temperatures, fuel geometries and the uranium and thorium-uranium fuel systems.

4. Structural materials

Various structural materials will be also studied. Irradiations will be carried out only in coolant channels used as test holes.

5. Design Studies

Fluxes, temperatures, etc. will be measured in the SRE to study performance characteristics.

6. Components

Various designs of components such as pumps, instrument chambers and etc. will be tested in the program.

The reactor will be shut down for all changes affecting the reactor core. It will be impossible by virtue of interfering mechanical latches to gain access to the core without first moving all control rods to the "in" position and tripping the safety mechanism. Any attempt to bypass this interlock will relieve the pressure in the reactor, and a scram will result.

In operating the reactor, the instruments of primary concern to the operator will be:

(a) Neutron flux level and period in the reactor
(b) Hot and cold leg temperatures in the primary and secondary loops of the heat removal system
(c) Sodium flow rates in the primary and secondary loops of the heat removal system
(d) Ambient and exit air temperatures at the air-blast heat exchanger

To achieve steady state operation at a desired power level, the operator will manually carry out an iterative process of adjustment of reactor flux, sodium coolant flow and air flow.

Before the reactor is brought critical, the sodium in the main primary and secondary loops will be brought up to 50 per cent of full flow. It will be held at a temperature of close to 250° F by the pipe heating system. After the reactor becomes critical, its power level will be raised gradually until a temperature differential is established across the reactor and the heating of the sodium is discontinued. At this point the coolant flow may be reduced below 50 per cent if desired. Then the reactor power will be raised slowly by adjusting the control rods manually and adjusting the inlet vents on the air-blast heat exchangers to obtain the desired operating conditions.

At power levels above 10 per cent of full reactor power, reactor outlet temperature in the primary loop will be held at 960° F, the flow rate of the sodium being manually set at the proper value to achieve this condition. As the power level is shifted from one steady-stage value to another the desired sodium temperature conditions will be maintained by changing the inlet vent settings and by varying the speed of the fans and the pumps.

A servo system will be provided to drive one rod in such a manner as to hold the neutron flux constant. This servo will be brought into use only after the desired flux level has been attained under manual control. A pump control system will be used to hold constant the speed of both the main primary coolant pump and the main secondary coolant pump at selected, fixed values after desired flow conditions have been attained manually. The pump controls may also be used to adjust the pump speed to hold some other variable in the system constant, (i.e., other than pump speed) such as the primary cold leg temperature.

The emergency heat removal system will operate continuously. The temperatures in the primary system will be fixed by the main primary loop, since the sodium streams are common in the reactor. The pump in the primary loop of the emergency system will operate at either full flow or 1/2 flow at all times. The pump in the secondary loop of the emergency system will have a variable speed drive so that the coolant flow rate may be set at the proper value to maintain the desired temperatures in the secondary of the emergency system. The air blast exchanger will have adjustable values and a variable speed for drive for air flow control similar to that provided for the main sodium-to-air exchangers.

VI. HAZARD EVALUATION

A. General

Nuclear reactors are similar to conventional pieces of power equipment in that their heat capacity between the operating temperature and the melting point of the materials of construction is small in comparison with the energy flowing from the machine in each minute of full power operation. Thus, if the flow of energy into a furnace or turbine is continued while the flow out of the boiler or generator is suddenly interrupted, the equipment will be reduced to a molten mass of brick, copper, and steel in a matter of minutes. That such accidents occur infrequently is ample proof of the adequacy and reliability of the safety devices used with them.

Nuclear reactors differ from the more conventional items of power equipment in three important respects:

1. The energy release rate may increase rapidly and may exceed the designed, full-power value by virtue of the chain-reacting nature of the system.
2. The energy release rate may not be reduced to zero in case of an emergency; the "afterglow" is an appreciable fraction of full power for a long time after shutdown.
3. With a high energy release rate is associated a high activity of fission products which are, in certain instances, highly toxic to the human system.

The radioactive fission fragments and plutonium can be potential hazard to the public only after they are released from the reactor. In the normal operation of the SRE most of the activity is restricted to the fuel, metallic uranium or thorium, and in general all the activity of great toxicity is confined within the cladding of the fuel elements. For radioactivity to become a public hazard there must be (1) a melting of the fuel elements to release the fission fragments and plutonium from the fuel, that is, the heat generation and heat removal must be out of balance, and (2) some mechanism to release the coolant and dissolved or suspended fission fragments and plutonium into the atmosphere or into ground water.

B. Nuclear Runaway

We have been especially interested, therefore, in restricting the nuclear energy which can be released under any circumstances to amounts which can be removed safely by the normal full-power operation of the cooling system. And we have been equally interested in making available temporary storage for the afterglow heat released immediately after shutdown and in providing a path by which this energy may, in time, be safely dissipated to the atmosphere by natural means.

We have analyzed a nuclear runaway during start-up. This appears to be the most serious, realistic accident in which the rate of release of energy exceeds the capacity of the cooling system. It is the most serious because the flux period may be quite short when the flux reaches the power range (above 1 kw); it is realistic inasmuch as start-up is a normal operation involving the deliberate and purposeful release of reactivity.

The reactor and cooling system will be protected by interlocks and control circuitry so that the control rods may not be withdrawn unless a minimum signal is obtained in the neutron flux sensing circuits; unless the safety elements are cocked and unless the cooling system is operating above 50 per cent of full power flow in all four loops. The rate of release of reactivity will be limited to a maximum of 0.3 cents/second; this will correspond to all four control rods being withdrawn at a maximum rate set by the synchronous speed of the motors and the associated positive gear train. When the flux is many decades below the power range, flux period circuits will become sensitive and will interrupt control rod withdrawal if the period becomes dangerously short. Adjustable flux level "set-backs" and "scrams" will also be available in this range. A fixed, flux-level "scram" will be set to operate at 125 per cent of full power flux. When the heat released in the fuel results in a sensible increase in the temperature of the coolant, high temperature "scrams" will become effective.

If all safety circuits fail and if control rod withdrawal is continued at the maximum possible rate, but if the coolant is flowing at the rate corresponding to full-power, our calculations show that the flux rise will be checked and reversed by the inherent negative temperature coefficient of reactivity of the system. The fuel will not melt during the transient, and probably not for a matter of several minutes thereafter. For a rate of reactivity release somewhat greater, 0.5 cents/second, the start-up time will be shorter, nearer 30 minutes instead of 50 minutes; but in the run-away the fuel would enter the gamma melt phase approximately 14 seconds after full power is reached. In both cases full power flux is reached and a temperature rise of coolant is signaled before prompt critical is reached.

During the runaway at start-up with a rate of reactivity increase of 0.3 cents/second if the coolant is not flowing, our calculations indicate that the fuel will melt approximately 6 seconds after full power is reached.

It is our conclusion that under the above conditions (U) a start-up runaway (with coolant flow) which results in melting of the fuel is ~~highly unlikely~~ will not occur. (U) An incident in which some fraction of the fuel does melt must however be recognized as ~~realistic and possible.~~ having a finite possibility in the event of failure of coolant flow during start up & injection of control circuit

C. Cooling System Failures

We have considered the possibility of failures in the cooling system which might weaken or destroy its ability to reject the reactor heat to the atmosphere. The system itself has been designed for highest reliability while the interlocking and safety instrumentation will be arranged to shut down or "scram" the reactor in case of failure of any component. As noted above, two cooling circuits are provided, a 20,000 KW circuit and a 1,000 KW ~~emergency~~ auxiliary circuit. Heat transfer is normally by forced convection, the ~~emergency~~ aux circuit is equipped with an emergency as well as a normal source of power. Each circuit may function at approximately 1 per cent of capacity by natural convection alone; this amounts to 200 KW for the main circuit.

To operate either by forced or natural convection there must be coolant in the loops. The primary or radioactive sodium loops which are joined at the reactor will be so sized and the pumps, exchangers, and piping so disposed that no single leak will interrupt flow of coolant by natural convection.

The cooling system will be interlocked with the reactor control to scram the reactor in case of failure of any component. In the case of a leak, loss of coolant will be detected first by a change in the pressure of the blanketing gas and second by a lowering in the sodium level in the reactor or in the surge tanks. Leak detectors on piping and small vessels and leak detectors located in the galleries will indicate presence of a leak somewhat later. Finally, loss of coolant would be [illegible] signal from the flow meters.

As described above, the reactor tank is surrounded by an outer container. The space between the tank and container is monitored to insure the integrity of both vessels; a signal indicating change in pressure of the monitoring gas or the presence of sodium in this space will be used to shut down the reactor. The volume of the space between the tank and container and the total volume of sodium in the primary system are such as to insure continued operation of the primary loop by natural convection, in the event of tank rupture.

D. Dissipation of Afterglow Heat

The worst realistic accident which we imagine is a break in the coolant discharge lines from the reactor which in effect separates the reactor from its cooling system. We postulate that the reactor is shut down at the same time the break occurs. Under these conditions our preliminary analysis shows that the afterglow heat released in a matter of seconds after shutdown will be absorbed and stored in the fuel and cladding with an accompanying temperature rise. We are aided in this by the moderately low average operating temperature of the fuel, approximately 950° F at the point of highest power density, by the high melting point of the fuel, and the high boiling point of the coolant.

In a matter of minutes the mass of moderator in the cell associated with each fuel element becomes effective as a sink for the afterglow heat. During normal operation the average temperature of the graphite is not greatly different from that of the fuel since the cell is bathed in coolant. The graphite therefor has a large capacity for afterglow heat. The operating temperature however, is high enough to minimize deleterious effect of radiation damage on the thermal conductivity.

For a matter of hours after shutdown the reflector core tank and thermal shield become effective as a sink for afterglow heat. The mechanisms redistributing heat within the reactor are conduction, thermal convection of the coolant between moderator elements and possibly local boiling of sodium in the fuel channels. All the sodium vapor formed is rapidly condensed in the coolant above the core.

Finally, in a matter of days, the biological shield cooling system comes into play to establish a steady state condition in which the afterglow heat is removed as rapidly as it is generated.

If the shield cooling system is inoperative the biological shield will absorb a certain amount of the afterglow heat. Heat will also be conducted at some rate through the biological shield into the earth from which it is finally dissipated at the surface to the atmosphere. The effectiveness of these mechanisms for heat storage and removal is to be evaluated quantitatively in subsequent analyses. In a matter of days after an incident it would appear possible to remove a substantial number of fuel elements and thereby reduce the total heat generation within the reactor.

It is our conclusion that in the case of the worst probable accident to the cooling system, shearing of the coolant lines, afterglow heat will not melt the fuel elements. If the reactor is scrammed immediately after the accident occurs, we do not believe it realistic to assume that the fuel will melt.

E. Worst Hypothetical Incident

Although the probability is very remote, an incident can occur during which a portion of the fuel will melt. This provides a mechanism whereby the fission products and accumulated plutonium may be released from the fuel and dispersed in the primary coolant. It is, therefore, proper to consider what sequence of events might realistically follow.

We cannot imagine a mechanism which would, in a short period of time, inject the activity into the atmosphere. The materials of construction cannot interact chemically with each other energetically. There will be no large store of energy in the graphite moderator arising from radiation damage which can be released with explosive violence.

The worst hypothetical incident is one in which a nuclear runaway during start-up is followed by a sodium fire which results in the dispersal of all the fission fragments and plutonium into the atmosphere in a period of 12 hours. Twelve hours represents a minimum time for use of the sodium based on considerations of availability of oxygen from the atmosphere. We consider three possible hazards resulting from the dispersal of airborne activity:

(1) A whole-body gamma and beta dose from a cloud passing overhead
(2) A beta dose resulting from inhalation of airborne activity
(3) The alpha and beta hazard consequent to a rainout over the Chatsworth reservoir and subsequent ingestion of water over a 10 day period

1. Cloud Activity - The gamma and beta dose from a cloud was computed according to the method set forth in WASH-3 except that all fission products were assumed to go into the cloud, the cloud thickness was taken as 1/2 the width, and a 2 mile per hour wind velocity was used. The results of these calculations gave a dose of 30 roentgens at a distance of 5 miles from the site, 7 roentgens at 10 miles, and 1.5 roentgens at 20 miles. The first figure is to be compared with 25-50 r, the range of allowable dose for a single exposure in a week's time or less if the level of 0.3 r/week (AEC tolerance) is not exceeded in the future. In evaluating the other figures it should be pointed out that 4 r in any 3 month period is satisfactory for "industrial hazards" on a continuous basis.

It should be noted that under the assumed conditions, 2 1/2 hours are available for warning persons at a distance of 5 miles, and 5 hours for warning persons at a distance of 10 miles. It is noted earlier that the population within 5 miles of the site is 3,000. Within a 10 mile radius the total population is approximately 65,000.

2. Inhalation Hazard - The beta dose from inhalation is computed on the basis of a 12 hour exposure. Separate calculations were made for the dose to the lung, thyroid, and skeleton. The activity (microcuries per cubic foot of air) used in computing these dosages are calculated by applying the results of the fluorescent aerosol studies to the diffusion of the fission products. This was done on a rate basis, so many curies released from the reactor in one minute being related to so many fluorescent particles released per minute. As noted above, 405 air samples were taken in the valleys below the site and the number of particles per cubic foot of air determined. The largest value obtained, 15 particles per cubic foot, was multiplied by the activity per particle (approximately 2.4 μc/particle) to obtain 36 $\mu c/ft^3$. With an allowance for decay with time the activity becomes roughly 10 $\mu c/ft^3$ ($3. \times 10^{-4}$ $\mu c/cc$) gross beta activity.

The figure of 10 $\mu c/ft^3$ is to be compared with 35 $\mu c/ft^3$ which we calculate to be the gross fission product beta activity corresponding to an assumed permissible emergency lung dose of 1000 rep in 60 days. 40 $\mu c/ft^3$

is the gross beta activity figure corresponding to an assumed permissible emergency thyroid dose of 1000 rep. Considering a skeletal dose of 100 rep in 60 days as permissible under emergency conditions, the activity should not exceed 100 $\mu c/ft^3$.

The figure of 10 $\mu c/ft^3$ may be compared also with ~~0.3~~ .03 $\mu c/ft^3$ ($10^{-6} \mu c/cc$) ? over a 24-hour period, the tolerance accepted by AEC for "bursts".

These data show that in the worst hypothetical incident the dosage calculated from available experimental data on diffusion is less than our assumed permissible emergency doses by a factor of between 3 and 10; the activity, on the other hand, is 300 times larger than the AEC value for "bursts", which applies to any populated area. It should be noted that the high readings (particles per cubic foot of air) were obtained under atmospheric conditions unfavorable to diffusion and that such unfavorable conditions may be expected to exist only perhaps 7 per cent of the time. It is interesting to note that the maximum count of 15 particles/ft^3 was observed at 15 miles from the site and represents a concentration 300 times that for "ideal diffusion" defined as perfect mixing in a cylindrical volume 15 miles in radius and 3000 feet in height.

3. Rainout over Chatsworth Reservoir - The hazard resulting from ingestion of fission fragments following rainout over the Chatsworth reservoir was calculated on the basis of a 3 miles per hour wind in the direction of the reservoir, precipitation beginning while the cloud is passing over the reservoir, and complete washout of the activity then over the reservoir. This results in something less than 4 per cent of the gross activity being precipitated in the reservoir. We assume further that the reservoir is full (10,000 acre feet of water) and that the activity becomes uniformly distributed.

The resulting activity due to strontium-89 is $4 \times 10^{-2} \mu c/cc$. This is to be compared with the Division of Biology and Medicine's proposed figure for an acceptable risk for a 10 day ingestion period of $9 \times 10^{-2} \mu c/cc$. The corresponding figures for plutonium are $10^{-6} \mu c/cc$ in the reservoir, and the proposed acceptable risk figure for a 10 day ingestion period of $5 \times 10^{-3} \mu c/cc$.

It should be noted also that the assumed conditions of wind direction and precipitation may be expected to exist only 6 hours out of the year. That is to say, in case of the worst hypothetical incident the probability of rainout in the reservoir is less than 0.1 per cent.

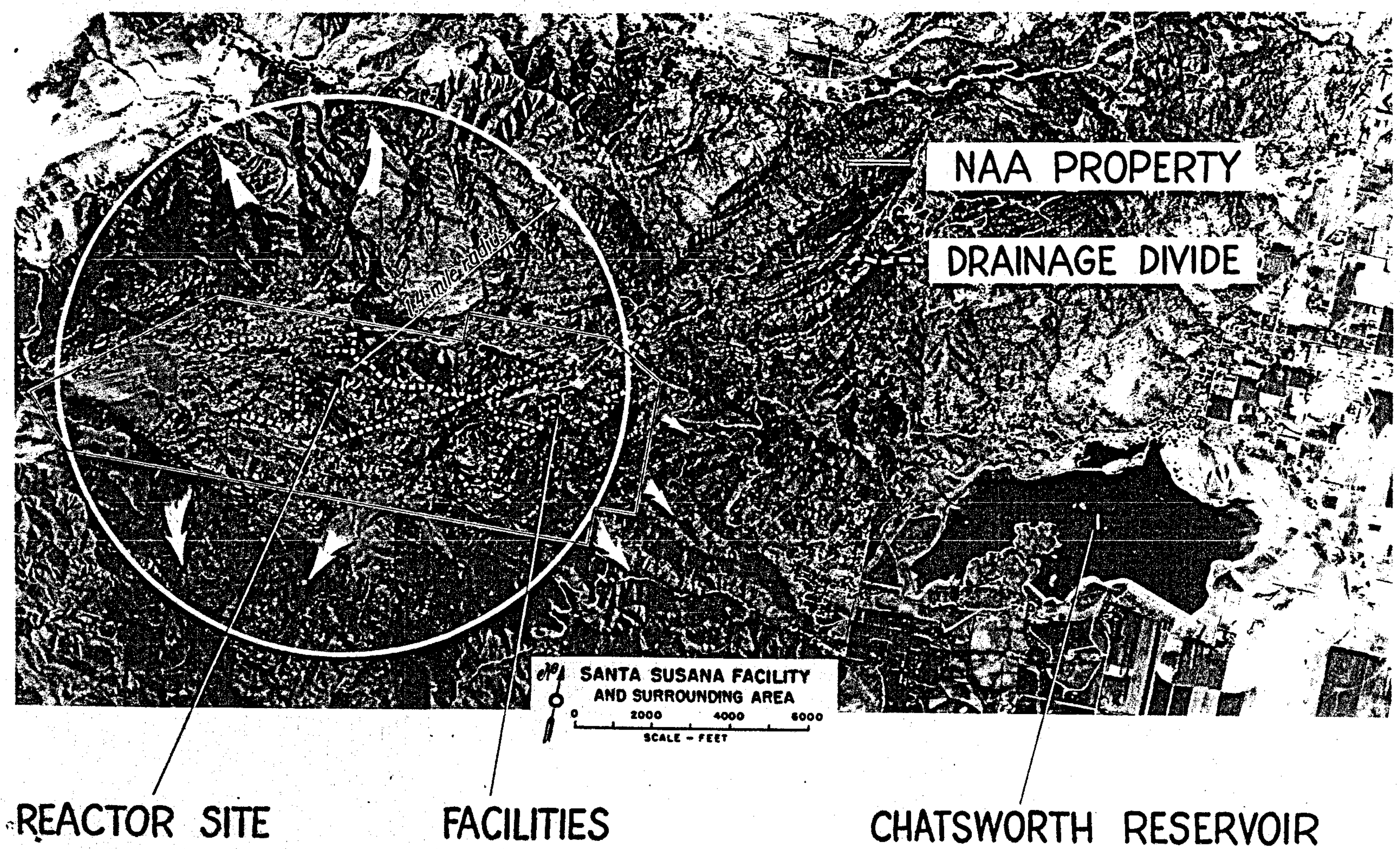
4. Discussion - The assumption that all fission fragments and plutonium contained in the reactor escape and are dispersed is unrealistic and pessimistic in the extreme. This assumption was made above, however, in preference to taking some arbitrary attenuation factor. It is considered best to give a value to this factor at the same time that the factors are assigned to attenuation by diffusion, probability of rainout, probability of existence of certain unfavorable conditions, etc.

Many factors will make the possibility of a fire very remote and will contribute to the reduction in the total activity escaping. Some of these factors are:

(1) Inert gas atmosphere in the compartments containing the primary sodium
(2) Five feet thick biological shield blocks normally covering the heat exchanger compartments when the reactor contains fuel
(3) Availability of fire protection equipment
(4) Only a small fraction of the fuel may be expected to melt in the incident
(5) Some of the fission products will be retained within the fuel material even though molten
(6) Not more than 20 per cent of the fission products are gaseous; the balance of the fission fragments and their products of combustion are solids at ordinary temperatures
(7) The products resulting from the combustion and hydrolysis of sodium, sodium oxide and sodium hydride, are also solid

We conclude that the SRE installation at Santa Susana presents no serious hazard to the public.

AREA SURROUNDING SRE SITE



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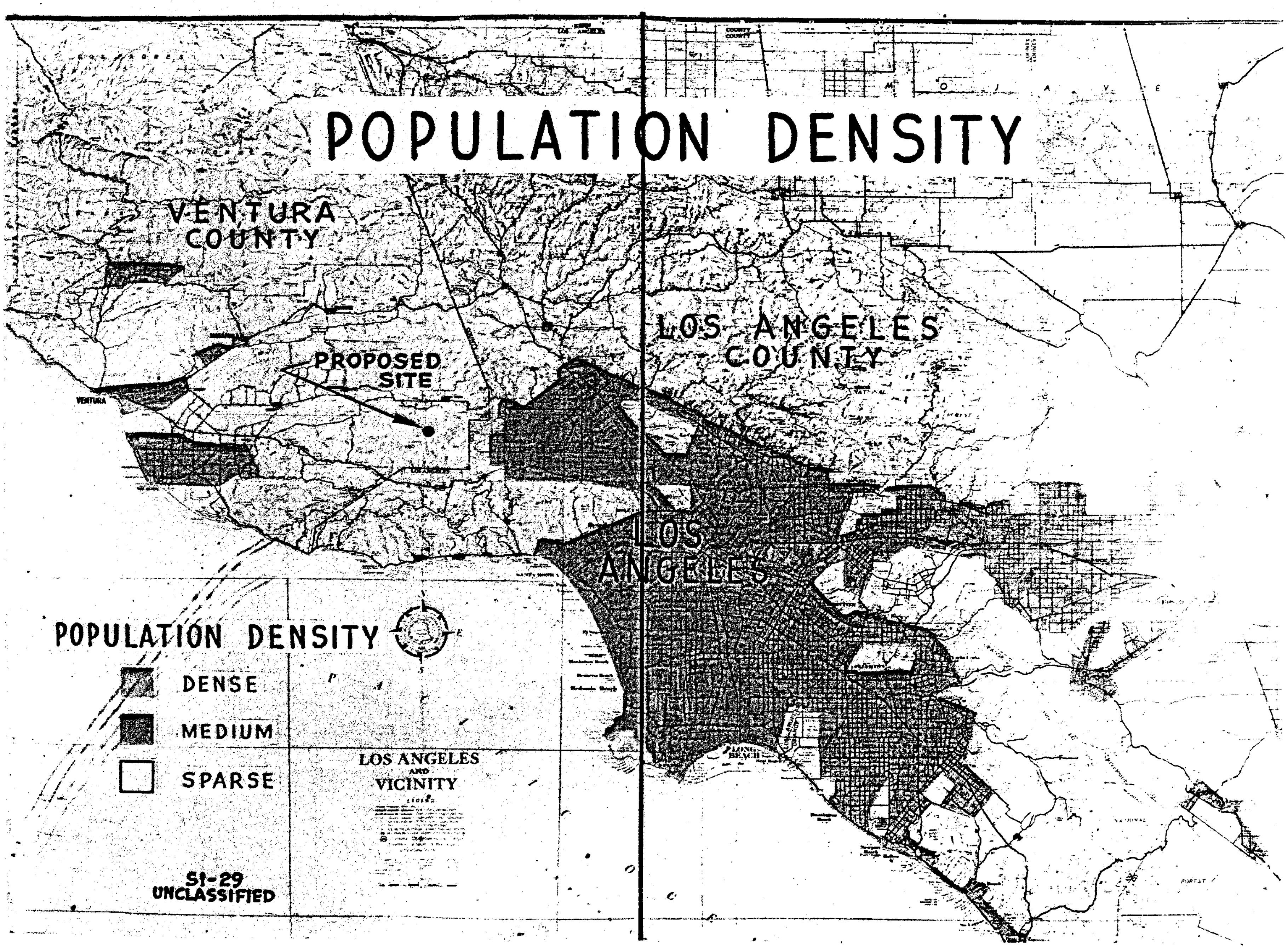

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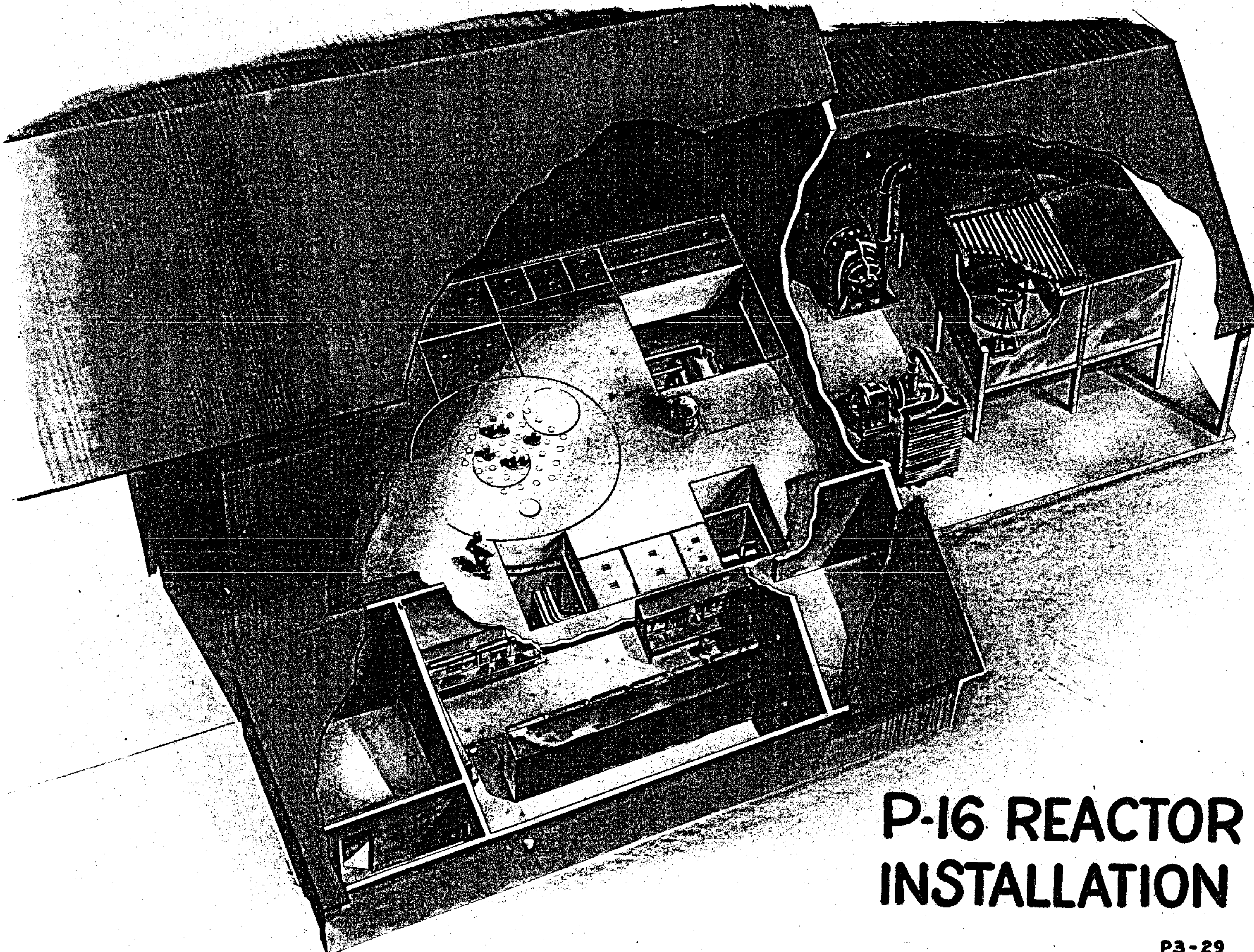

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