Part II

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M E M O R A N D U M

THEORETICAL CONSEQUENCES OF HYPOTHETICAL ACCIDENTS TO LARGE NUCLEAR POWER PLANTS

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In late 1964, at the request of the Atomic Energy Commission, a group of scientists and engineers drawn principally from Brookhaven National Laboratory was asked to perform a review related to the safety of nuclear reactors, not from the point of view of the standard safety analysis of a nuclear reactor, but going beyond the usual limits of analysis and considering effects of circumstances that could be termed incredible or perhaps practical impossibilities. The assigned task was to disregard questions of probability of accidents and to consider only consequences. The results of this review, which brings up to date a similar one performed by a Brookhaven group in 1957, are summarized briefly here.

Typical Reactor

The reactor taken for consideration is assumed to generate 3200 megawatts thermal power and to have operated continuously for a complete fuel reloading cycle of 1000 days. The fuel is typically stainless steel clad uranium oxide, and the coolant is water. The reactor is shielded with concrete and is situated in a containment building which is leak tight to less than 1% of the contained gas per day and is able to withstand large internal pressures.

Typical Sites

Two types of reactor sites were considered in this study. The first is a typical present-day site with a neighboring population distribution that does not differ significantly from that of

the typical 1957 site. It is expected that during the next decade reactors may be built in urban load centers. Thus the second site treated in the analysis was in the center of an urban area.

Description of Accidents Selected for Study

Nearly all of the accidents that can be visualized as possible would be of the nature of malfunctions that would interfere with operability of the reactor but would not affect the public. For an accident to become a hazard to the public, there would have to be a sequence of events that in some fashion allowed the fission products normally fixed within the fuel elements to escape the boundaries of the fuel elements as well as a number of other barriers.

In most reactor safety analyses, this sequence is assumed to be started by a catastrophic rupture of a primary coolant system pipe. If a severe break of this kind were to occur, it would allow the primary coolant to escape from the reactor vessel and could lead to inadequate cooling of the fuel elements if they indeed did become exposed to air. Without an occurrence of this or a similar nature, there would be no possibility of liberating fission products and no hazard to the public. Therefore, the same initiating event, a catastrophic rupture of a major primary system coolant pipe, is assumed in the present study. Up to a point, the assumptions on the course of the hypothetical accident closely follow those in

conventional safety analyses. However, to achieve the purpose of the present study, which involves identifying the point at which damages and public injury would occur, it is necessary to suppose that all of the means of assuring safety have failed to function. For instance, the emergency cooling system is simply supposed not to operate as it should. In addition it must be assumed that some large penetration in the reactor containment building is open at the time of the accident, or that the containment building is damaged by a missile from the accident, so that containment is violated. At this point, the hypothetical accident would become a hazard, and its consequences would be severe. In order to identify the point in the spectrum of hypothetical reactor accidents where public injury and financial damage would begin, it has been necessary to assume that a very improbable event is followed by a failure of a complete set of safeguards that are engineered to prevent hazards to the public or to reduce these hazards.

Fission products would be liberated from the fuel when it melts some time after the hypothetical accident is started by the loss of coolant. The means by which the fission products are freed from the fuel, diffuse out of the reactor vessel, and disperse throughout the atmosphere of the building have been analyzed by techniques of heat transfer and thermal convection, the chemistry of the fission products, and the physical characteristics of the fumes or aerosols formed by these materials.

It is not possible to identify any one point in the primary piping circuit as more subject to the improbable mode of failure that has been assumed than are other points. Nor is it possible to make general assumptions on the location of the failure relative to the complicated network of pipes, ducts, electrical conduit, passageways, etc., that is commonly characteristic of the primary piping region. Thus it is not possible to estimate the effect of this geometrical question on the nature or amount of fission products that would be released to the containment building. The analysis has therefore been based on the additional assumption of instantaneous uniform mixing of fission products in the containment building atmosphere as they are released from the damaged reactor. The effect is that the consequences of the hypothetical accident are overestimated.

Three cases have been analyzed: Case I - Complete Containment, Case II - 0.2% Per Day Leakage Rate, and Case III - Breached Containment. For the purposes of this study, the following model was used to approximate the course of the hypothetical accident and to obtain results for cases I, II and III. It was assumed that for a time after the initial pipe rupture, water is being boiled out of the reactor vessel and the fuel remains intact. After the water is gone, the temperature of the fuel begins to rise. Eventually the fuel melts and drops to the bottom of the vessel. From a computation of the total radioactive decay heat generated by the

fission products in the first six hours after shutdown, it was found that there was enough heat available to melt the fuel cladding and 95% of the fuel itself, to free the releasable fission products, and to melt through the bottom of the reactor vessel. Knowledge of the distribution of decay heat in the fuel leads to an estimate of the time dependence of melting of the fuel.

It was assumed that at the time of fuel melting, the atmosphere within the reactor vessel is a mixture of steam and air. Experimental data on release of fission product isotopes from molten uranium oxide in air and steam were used to estimate the fission product release. The predominant particle size of the aerosol that is formed by agglomeration as fission products are evolved from the melt is calculated to be approximately one micron.

Deposition and retention of the fission product aerosol within the reactor vessel and in the primary piping region could not be estimated, because the factors which bring about aerosol deposition, i.e. obstructions to air flow and the nature and extent of surfaces available for deposition, cannot be specified for a generalized case. If it is assumed that flow is limited only by the size of the ruptured pipe, the calculated flow rate is large enough to sweep the aerosol out of the vessel and into the containment building almost as rapidly as the aerosol is evolved from the melt. This additional conservatism causes the calculated effects

to be greater than those which are truly representative of this simplified model of the hypothetical accident. But no way to avoid this conservatism is possible in general.

Radiation Effects

The pattern of radiation exposure dose as a function of time of decay of the fission products was then calculated. The analysis done took account of the individual fission products that would be released if the fuel elements were to melt, and the values of radiation dose in four days per microcurie inhaled, for each isotope and each critical organ. From the total exposure pattern, it was found that the dose to the gastrointestinal tract was a controlling factor in determining exposure levels for categories of no injury, injury unlikely, illness, and lethality. It was found that time of inhalation had little effect on the relative exposure to the various organs.

Ground contamination effects were not examined as extensively as direct exposure effects. Federal Radiation Council action limits have been set for I^{131}, and are imminent for Sr^{90}. Corresponding criteria could be deduced for gound contamination levels which would require destruction of crops and those which would place long-term restrictions on agriculture from the remaining isotopes. However, for the purposes of this study, indirect exposure effects

from I^{131} and Sr^{90} are sufficient to illustrate the nature of contamination from the hypothetical accident that would be a hazard to the public.

Atmospheric Dispersion

The meteorological analysis in the 1957 study was known to be incomplete and unrepresentative in several respects. Some of the difficulties stemmed from a lack of technical knowledge, but inability to cope with the immense number of calculations required for an adequate representation also limited the earlier attempt.

From a technical point of view, the most important new meteorological knowledge incorporated into the present evaluation reflects a better understanding of dispersion conditions over urban areas. Most earlier meteorological evaluations have assumed that the extremely stable condition typical of open country at night can exist normally over urban areas as well. Data obtained from several sources in the last few years have shown that such a condition over a city is at best extremely rare and that nocturnal conditions over the city are usually characterized by weak turbulence in the lowest layer, capped by a stable layer aloft.

Several other refinements have also been made. The most important of these include recognition that dispersion does not continue indefinitely in the vertical, but it is always restricted by an atmospheric lid at a height ranging from a hundred to a

thousand meters. Similarly, this study has incorporated the effect of the change in wind speed with height, which is known to be a consistent feature of the lower atmosphere. Finally, while the original study assumed that the meteorological conditions existing at the time of the assumed accident continued indefinitely without change, the new computation permits complete flexibility in change of the conditions as a function of either time or distance.

Diffusion at great distances is not understood much better than it was seven years ago. It is partly for this reason, as well as the fact that dispersion would often be over water after 1000 km, that this distance has been fixed as the outer limit of the calculations.

Calculational Technique

The features which have been briefly described in the foregoing sections were incorporated into a digital computer code with which radiation exposures and ground contamination were calculated for a given set of input data. These data were the release of fission products as a function of the time after release, the meteorological conditions prevailing at that time, and the type of site. The radiation effects in the area surrounding a country or urban site were obtained. The results took into account the day/night variation of meteorology characteristic of the geographical area considered.

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Discussion of Results

The task assigned in the present review was to disregard questions of probability of accidents to reactors, and to reevaluate only the effects of hypothetical accidents. This reevaluation was to take into account the increased technical knowledge obtained in the past seven years, as a result of continuing research in reactor technology and associated fields. Furthermore, this study was to go beyond the usual limits of safety analysis, to make clear what the safety features of reactors guard against, and to identify the point in the spectrum of hypothetical reactor accidents where public injury would begin. In order to fulfill this objective it has been necessary to suppose that all of the means of assuring safety have failed to function.

For Case I - Complete Containment, in which no radioactive fission products were released to the atmosphere, estimates of injuries from radiation exposure were very low and there were no hazards from ground contamination.

In Case II, the assumption that some leakage of fission products takes place makes the meteorology of some importance. Thus, if the hypothetical accident took place under daytime weather conditions, with relatively good atmospheric dispersion, calculations imply that some injuries from radiation exposure would result, but no lethal doses would be sustained. Under nocturnal inversion

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conditions the hypothetical accident would cause many more injuries and there would be a number of lethal exposures.

Areas contaminated in excess of FRC limits for I^{131} would range from 1000 to 3000 square kilometers, depending on the weather conditions at the time of the release. It should be noted that the assumption that leakage continues at a constant 0.2% per day of contained air mass is unrealistic, because heat exchange to the environment and associated changes in driving pressure have been neglected.

It is, however, in Case III that the study has been extended to a point commonly considered incredible or a practical impossibility. Here it was assumed that fission products were liberated to the environment as soon as they were evolved from the melting fuel. The implications of this part of the study, which perhaps should be regarded as a mathematical exercise, are that considerable illness, lethal exposure, and land contamination would occur for all weather conditions and reactor sites considered.

An interesting sub-case of case III we have called the "hot puff" release. It is assumed here that all of the airborne fission products are released at the same time as the primary system ruptures, and that they are therefore retained in the hot cloud of steam that was generated. It was found that the temperature of the released cloud itself, with heat added by decay of the radioactive fission products, would carry the material into the upper atmo-

sphere, where it would be harmlessly dispersed.

The study also indicated that for any given assumptions on the rate at which heat is lost to the containment structure and its contents, and eventually to the environment, there is a power level beyond which engineered safeguards for additional heat removal become more necessary. For systems typical of those studied, and for a hypothetical accident of the size considered, this point is reached at about 500 megawatts of thermal power. Beyond the critical power level, the heat from decay of radioactive fission products would in time raise the internal pressure above the strength capacity of the containment building. Reliable heat removal would transform some hypothetical accidents from the category of Case III to levels considerably milder than case II.

It is appropriate here to compare the results of the present reanalysis with those of the 1957 study. At the outset of the comparison, it should be stated that note has been taken that considerable improvement in numbers of engineered safeguards and their reliability has occurred in the intervening period. The effect of this trend can only be to reduce even further the probability of reactor accidents, and to lower the consequences of any failures. However, it was made clear in the assignment of the present task that questions of probability and, thus even reductions in probability, were not to be considered. Therefore, we have as in 1957

assumed that all safeguards, such as emergency cooling, fission product retention devices, containment, etc., fail to perform their intended functions.

We have used more reliable knowledge and calculational techniques to improve formulae, values of parameters, and data-handling, to arrive at more accurate and more detailed calculations.

This has been particularly true for fission product releases from molten fuel, atmospheric dispersion of airborne materials, and the processing of calculations with modern computing machines. In total, these particular refinements have only improved the accuracy of prediction of the effects of the extreme hypothetical accidents considered, and they have had little effect on the size of these estimates. The one real change introduced by the reassessment is a result of the increase in size of the reference reactor. Because reactors now being contemplated are several times larger than those in prospect in 1957, and fuel cycles are longer, it is an inescapable conclusion that, assuming the same kind of hypothetical accidents as those considered in the 1957 study, the theoretically calculated damages would not be less, and under some circumstances would be substantially more, than the consequences reported in the earlier study.

For very large accidents, such as those considered in case III, the total damage to people and property under the worst weather

conditions would be substantially the same, whether the reactor is in a large city or some distance away, assuming that in the latter case the wind is directed toward the city in the time period considered. The advantage of a "country" location for a large reactor is related to less severe accidents than the extreme ones of case III. A smaller accident would, of course, be less hazardous at a more remote site.

The present report lists only the conclusions of the restudy, and has been issued now in the interest of early dissemination of these conclusions. Details of the calculational techniques and the values of important parameters of interest to the nuclear community will be published as separate reports in the technical literature.

RETURN TO
OFFICE OF THE SECRETARY
ATOMIC ENERGY COMMISSION

BRIEFING FOR THE PRESIDENT
ON POTENTIAL HAZARDS OF ATOMIC REACTORS

On January 29 the President received a briefing on reactor hazards. The following representatives of the Atomic Energy Commission participated in the briefing:

Chairman John A. McCone

Dwight A. Ink, Assistant to the Chairman

H. L. Price, Director, Division of Licensing & Regulation

Dr. Clifford K. Beck, Chief, Reactor Hazards Evaluation Branch

I - SUMMARY

1. Dr. Beck started the presentation by outlining the nature of the hazards, reviewing past accidents, and identifying the possible types of accidents and safeguard measures taken. He explained the need for isolation and containment and other safety measures and mentioned the reviews made within the Commission and by the Advisory Committee on Reactor Safegaurds. He mentioned the special safety problems involved in the maritime program and the aircraft program, and briefly discussed the possible range of consequences of major reactor accidents based on the Brookhaven study of 1957. He concluded with a brief description of emergency procedures which have been established for dealing with radiation incidents. A statement of the substance of the briefing is attached.

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2. Mr. Price stated that the study of possible consequences of reactor accidents was made for the Joint Committee of Congress in connection with its consideration of indemnity legislation. It dealt with the theoretical possibilities of major catastrophes. It did not attempt to describe expected reactor accident experience.

3. Chairman McCone emphasized the safety review of all reactor projects within the Commission by a separate staff organization having no responsibility for construction or operation of reactors, and the additional review by the statutory Advisory Committee on Reactor Safeguards, an independent group composed of part-time consultants to the Commission.

4. Following the presentation there were questions by the President and a discussion of various aspects of reactor safety.

II - POTENTIAL HAZARDS FROM POWER REACTORS

1. The Principal Hazard: Accidental Release of Radioactivity Inventory

The primary hazard of a reactor is the possibility that the radioactive fission products produced in the reactor might be released and dispersed into areas beyond the owner's control. Fission products are the fragments of uranium atoms, the "ashes" of the nuclear reactor, accumulated in the core of a reactor as the nuclear fuel is "burned". The magnitude of this fission product inventory is approximately in proportion to the power level at which the reactor is operated. If this material should be accidentally released from containing structures, dispersal to public areas could occur by way of the underground water table or nearby streams or through the atmosphere.

In a major reactor accident -- fire, internal explosion, or power runaway with consequent meltdown of fuel -- a portion of the fission product inventory may be released.

The factors of concern in such an incident would be:

a. The magnitude of the fission product inventory. This depends primarily on power levels.

b. The radioactive nature of the fission products. The average half-life of fission products in reactors is much longer than the fission

products of weapons. The average half-life increases as the fuel cycle is increased.

c. The physical nature of the fission products. Some are gases, such as xenon and krypton; some are volatile and chemically active, such as iodine; some are metals or non-volatile solids and would generally tend to remain in the fuel unless violently dispersed.

d. Condition of the airborne release. Whether the airbrone component of the released material is in gaseous form, is affixed to fine particulate matter (such as carbondust from the burning of graphite), or is physically attached to larger particles (e.g. dust from a steam or chemical explosion), has a major influence on the subsequent pattern of dispersion and deposition of this material.

e. Weather conditions. Atmospheric turbulence, temperature inversion, rain, and other meteorological factors have important effects on whether airborne radioactive materials are dispersed, channeled or deposited.

In all power reactors now built or under construction, the nuclear fuel in the reactor cores is in solid form and

is clad or sheathed in leak-tight metallic coatings. These coatings serve as an initial barrier against escape of fission products. A typical arrangement of a reactor facility is shown in Chart 1. A high pressure, high integrity vessel constituting a portion of the primary coolant system encloses the core and offers the second major barrier against release of fission products in case the fuel structures themselves should be accidentally disrupted. Massive shielding structures surround the reactor core and its containing vessel to protect nearby workers from radiation. These structures would also offer substantial resistance to the dispersal of radioactive materials. Finally, there is for most power reactors built near centers of population, some sort of external containment shell which serves as a barrier of last resort against the escape of any internally released materials.

Chart 2 is a recent photograph of the Dresden Nuclear Power Station, now nearing completion near Chicago. The containment vessel and the isolated environmental conditions are shown. Such safeguards as these and others, mentioned below, lead to the unanimous opinion among reactor experts, that the likelihood of a reactor accident which would release dangerous amounts of fission products beyond the boundaries of the facility is extremely low. How low this probability might be is impossible to know quantitatively, but rough guesses of 1 chance

in 100,000 to 1 in a billion per reactor - year have been made. This could be interpreted as the chance that there might be one major reactor accident per hundred years if 1000 reactors were in operation. No firm reliance can be placed on such estimations, but they do give an indication of the strongly held belief that major reactor catastrophies are extremely unlikely.

2. Past Accidents

No significant accidents have occurred thus far in this country in regularly operated power or production reactors. Two accidents have occurred in experimental reactors.

During an exploratory transient operation with the Experimental Breeder Reactor No. 1 in 1955, in which some of the protective devices had been deliberately disconnected in order to conduct the experiment, the power level rose to unintended levels, and partial meltdown of the core resulted. The fission product inventory was confined to the system and no one was injured.

In 1958, in an advanced design experimental reactor in the Aircraft Propulsion program, during initial trial operations at low power, a power flash-up occurred. This was due primarily to faulty design in the control and safety instrumentation system. Some of the fuel elements were partially melted; about 1% of the fission products was discharged through the air-cooling system to the atmosphere. Ground contamination outside the reactor building was slight; no one was injured.

At Windscale, England, in 1957, the only known serious accident with a steadily operated (production) reactor occurred. During a special operation with reduced cooling designed to relieve the accumulated radiation energy in the reactor's graphite by annealing, and because of inadequate instrumentation, the reactor became overheated and a portion of the graphite and fuel burned. A large portion of the fission products released from the burned fuel was swept by the cooling air up the stack and into the atmosphere. Filters at the top of the stack were able to retain a considerable portion of the particulate material, but gases and fine particulate matter were dispersed to the atmosphere and a portion was subsequently deposited on the countryside. The level of contamination was not sufficient to be a health hazard from direct exposure, but for about three weeks milk from cows in a narrow area about 20 miles long was withheld from human consumption.

3. Types of Accidents

In the above cases, release of fission products resulted from overheating or burning of the fuel. This would occur in almost all types of catastrophic accidents. Overheating or burning may result from many causes. Some of these are;

a. Loss of coolant or failure of the coolant circulating pumps may lead to fuel meltdown.

b. Excessive power levels may cause fuel burnout or meltdown.

c. In graphite or organic moderated reactors, the moderator may catch fire and overheat the fuel.

d. In some reactors, violent water-metal reactions may occur if the metal becomes overheated (aluminum-water; zirconium-water).

e. In reactors with high pressure coolant systems (pressurized, boiling water, etc.,) the stresses in the system may eventually lead to failure and consequent loss of water coolant.

There is no hazard of a bomb-like nuclear explosion in a reactor. Overheating could cause chemical or steam explosions, of a violence comparable to that possible in other industrial operations.

4. Safeguards; Containment, Location

In every reactor, great care is taken to provide safeguards against the particular kinds of accidents to which it might be vulnerable. These include emergency cooling systems against failures of the primary cooling system; automatic, fast acting shutdown mechanisms against excessive power surges; extraordinary care in assuring high integrity materials and high performance machinery.

The safety aspects of all reactor projects in this country, whether privately or Governmentally owned, including military projects, are reviewed by a technical group within the Commission which has no other responsibilities. The Reactor Hazards Evaluation Branch extensively reviews the design, containment, and location of each reactor and other considerations pertaining to safety.

In addition, all power and test reactors and large reactors of other types are reviewed by the statutory Advisory Committee on Reactor Safeguards, a group of part-time technical-expert consultants. The operation of Government-owned projects is subject, of course, to administrative supervision. The construction and operation of privately owned reactors are subject to detailed licensing requirements and are periodically inspected for compliance with all safety requirements.

Despite these precautions there still remains some uncertainty relating to the possibility of a serious reactor accident. Hence, the following two additional precautions are observed for most large reactors built near populated areas: 1) provision for some degree of external containment around the entire reactor facility, and 2) selection of a location where there is some degree of isolation immediately around the reactor and a relatively low population density in neighboring areas.

5. External containment may vary from high-integrity steel vessels of high pressure capacity and low-leakage specification,

to relatively open structures equipped with filtering-scrubbing devices in the ventilation exhaust system. The exhaust systems are designed to keep the structure at pressures less than atmospheric to prevent unfiltered outleakage in case of accident.

The purpose of containment vessels is to provide a barrier of last resort against release to the atmosphere of the fission product inventory in case of reactor disaster.

The value of an external containment vessel is difficult to assess. Its chief function is to provide insurance against the extremely unlikely catastrophic accident which would only occur if all internal safety devices and precautionary measures fail. If, in an instance of this kind, the containment vessel is successful in preventing major fission product dispersion, the protection afforded is well worth the cost. On the other hand, it is possible that the containment vessel might be left open just at the time when it is needed, and that accidents could occur which might breach the containment vessel. Thus, the containment vessel offers added protection, but it does not insure absolute protection.

6. Location of reactors within an exclusion zone, and in an area of low population density is a practice which arises from recognition that accidents could occur which would release some fission products from a reactor and even from a containment vessel, if one is present.

An exclusion distance of from about 1/4 to 1/2 mile has been provided for most large power reactors now under construction. The reasons for the desirability of such a zone are:

a. The movement of any persons present in this area can be controlled by the owner and hence exposure hazards in case of accident can be minimized.

b. In case of a release of radioactivity which is contained within the reactor facility direct radiation to persons outside the operator's premises would be minimized.

c. In case of small radioactivity releases to the atmosphere, intentional or otherwise, the exclusion zone assures some dilution before the radioactivity reaches public areas.

d. In case of major releases under the majority of atmospheric conditions, an exclusion zone would afford some period of time in which residents outside the area could be warned of the approaching hazards.

The specifications for a low population density in areas around a power reactor have not been precisely defined. However, it is considered desirable for a large reactor to be about 20 miles or so from a major city, and at least a few miles from

smaller towns, though particular features of individual facilities may in part offset this requirement.

This feature of location is desirable because:

a. In case of catastrophe, a low population density assures that serious exposure of large numbers of people is minimized.

b. In case of catastrophe, a relatively few people might be evacuated without radiation injury, while large numbers of people could not be evacuated.

c. In case of catastrophe, the economic value of property contaminated would tend to be less for areas of low population density than for areas of high population density.

7. The Maritime Problem

Marine vessels propelled by nuclear power present no different basic problem from that arising from any power reactor. Possible release of the radioactivity inventory constitutes the principal hazard. However, there is a high degree of probability that catastrophic accidents in marine vessels would be concomitant with submerging or flooding of the reactor. In many types of reactors, flooding would tend to prevent any release of fission products. Furthermore, any release that did occur would be into the water. Most of the fission products would be retained in the water, and

the hazard to people in nearby areas from airborne radioactivity would be correspondingly reduced.

It is difficult to make a general estimate of the damages which would be caused by heavy contamination of the waters of a harbor. So much depends on such factors as currents and tides, flushing times, activities in the harbor and the effects of contaminated waters on adjacent shore activities. In general, however, it is clear that the principal consequences of a flooded or submerged release would be economic. There would be an interference with harbor activities, and an indirect threat to health by the presence of contaminated water. The level of activity in the water and the quantity of airborne activity would hardly be high enough to pose a threat of serious direct radiation exposure to large numbers of people.

On the other hand, it is possible that reactor accidents in marine vessels could release fission products directly to the air. In such case, the threat of airborne hazard to people in adjacent areas would be no different from that arising from a land-based power plant. These and other special circumstances affecting the safety of marine vessels may be summarized as follows:

a. Marine vessels operate in port areas which are located immediately adjacent to population centers where stationary nuclear power plants would not be permitted. (A given vessel, however, is usually in a port area for only a small proportion of its operating time.)

b. The mobility and maneuvering requirements of a marine vehicle make it more susceptible to operational accidents than is the case for a steadily operated power reactor. The navigational hazards in port areas, where the consequences of accidents would be most serious, make accidents more probable here than out on the open sea.

c. In marine vehicles, particularly submarines, most catastrophic accidents would have as their most serious consequence contamination of the water. In a harbor area the seriousness of water contamination would depend on numerous factors previously mentioned.

d. In some marine accidents, airborne fission product releases could occur, i.e., in non-flooding types of accidents, which could offer serious hazards to inhabitants of adjacent areas.

e. For nuclear propelled merchant vessels, assurances of safety must be placed on the adequacy of containment and dependability of other safeguards, the quality of training and supervision of personnel, and on limitation and control of operations in costal areas.

f. It is the general opinion of persons familiar with the problems of marine vessels propelled by nuclear power that the operation of such vessels in coastal areas should continue to be on a limited and controlled basis until the real extent of potential hazards are better understood.

8. The Nuclear Aircraft Problem

As in other nuclear facilities, the fission product inventory in an aircraft reactor is proportional to the power level, and release and dispersion of this material into populated areas is the principal hazard. Operation of nuclear aircraft over populated areas would offer the possibility of crashes and fission product releases in the most hazardous locations. Restriction of flights to "corridors" having low population densities or to flight paths over extensive bodies of water would substantially reduce the potential hazard from accidents, but would not eliminate it.

Unique aspects of the nuclear aircraft problem are:

a. The usual safety implications of malfunction or misoperation of reactors are greatly intensified for nuclear aircraft because of the possibility of a crash of the vehicle.

b. An external containment vessel for the reactor facility cannot be provided in an aircraft. If one were provided, crash of the aircraft would

probably demolish the containment as well as the reactor, with resulting fission product release.

c. The operation of a nuclear aircraft in the air-space environment, with weight limitations on over-all shielding, and the high performance demand on components required, will impose the necessity of unusual operational controls.

d. These added aspects of hazard suggest that operation of nuclear powered aircraft should be carefully controlled, and limited to areas where a crash would not be within dangerous range of inhabited areas.

9. Consequences of Accidents in Power Reactors

A study has been made for the Commission at the request of the Joint Committee on Atomic Energy on the theoretical possibilities and consequences of catastrophic accidents in large nuclear power plants. A group of scientists directed by representatives of Brookhaven National Laboratory collaborated in making this study and the results were publicly released at the time they were sent to the Joint Committee on Atomic Energy. To make this study, arbitrary and uncertain assumptions had to be made on many of the factors having a vital effect on the magnitude

of the result. In all cases, the values which appeared most likely and the assumptions which appeared most reasonable were made.

The study was made for a 500 thermal megawatt power reactor in a typical location, i.e., 30 miles from a major city, with a 2,000 foot exclusion distance, and with reduced population density beginning at zero at the site boundary and increasing gradually with distance to a value at the distance of the city equivalent to that along the eastern seaboard.

From the variety of detailed situations studied, two are reported here:

a. The Contained Case.

It was assumed that a major accident occurred releasing the entire fission product of the inventory from the core, but into a high integrity containment vessel which permitted no leakage to the environment. Table I contains a summary of the results.

It should be noted that the only hazard from an accident of this kind to people beyond the confines of the plant would arise from the direct radiation streaming out from the containment vessel in all directions. This radiation decays in time and is attenuated with distance

so that beyond a few thousand feet dangerous exposures would not occur. People closer to the plant would have to be evacuated for a period of time to avoid injurious exposure. As indicated in the table, less than 75 people would have to be evacuated from the adjacent area of over about two square miles. If this evacuation could be accomplished within two hours no one would receive lethal or injurious exposure. If the evacuation were accomplished in 24 hours no lethal exposures would result but six people would be injured.

b. The 50% Release Case

For this accident, it was assumed that 50% of all fission products escaped into the atmosphere and was subsequently dispersed over the countryside. Table II contains a summary of the personal injuries which might result. It should be noted that an accident of this magnitude is larger than any accident having a credible likelihood of occurrence. It is difficult to see how even deliberate sabotage of a nuclear reactor could cause the release of this percentage of the fission product inventory. Thus the results

shown represent a magnitude of damage which it is believed will never be realized, though conceivably it could be realized.

In Table II, personal injuries which could occur as a result of this accident are shown under three combinations of conditions. If the accident should occur under the most favorable conditions no persons would be lethally exposed and none injured. On the other hand, as indicated in column 3, under the worst conditions which might exist, and these might be expected for as much as 5 or 10% of the time, as many as 3400 people might receive lethal exposures and more than 40,000 might be injured. The second column of this table indicated that for typical conditions which might be expected over 3/4 of the time, less than 10 people would be lethally exposed and less than 100 injured.

It should be noted that in addition to the above personal injuries which might result from a serious accident, there would be costly and hazardous land contamination over broad areas, thousands of square miles in some cases, which might require temporary or extended evacuation of persons and restrictions on various activities, particularly agricultural activities.

Though it is not anticipated that accidents having consequences as bad as the worst ones listed above have any credible likelihood of occurrence, it is almost certain that there will be accidents with reactors and that in some cases these could result in substantial injuries to people and contamination of land areas. We can hardly hope, with the increasing numbers of reactors now being built, despite the many precautions diligently applied, to escape entirely from the possibilities of serious accidents.

10. Emergency Procedures

To minimize the consequences of accidents if such should occur, it is required that every facility develop plans and procedures for handling accidents and for protecting the public in case of emergency. Usually this involves developing with local police, fire departments, civil defense and public health leaders an understanding of how matters would be handled in case of reactor emergency. These plans include procedures for obtaining outside assistance in case the emergency should exceed the capacity of the local facility.

Each AEC field office throughout the country has emergency procedures, teams and equipment ready for assistance in any

nuclear emergency which might occur within its area of jurisdiction. Eight of the AEC field offices, one in each of the eight regions which together cover the entire United States, has been designated as a regional emergency center. Additional competence for dealing with reactor accidents has been developed at each of these regional offices. Procedures have been established whereby field offices can call on military assistance and trained radiological teams available at various military centers as needed.

Finally, at the Washington Headquarters of the AEC, a Radiological Incident Center has been organized, with standby personnel appointed, to be manned at any time a major accident should occur. This center can coordinate assistance to any location and secure additional aid from military, civil defense and other agencies as may be needed.

TABLE I

THE CONTAINED CASE

(ASSUMING A 2,000 FOOT EXCLUSION DISTANCE)

	ASSUMING EVACUATION in 2 HOURS OF NEARBY AREAS	ASSUMING EVACUATION in 24 HOURS OF NEARBY AREAS
PERSONAL INJURY		
LETHAL EXPOSURE	0 PERSONS	0 PERSONS
INJURY LIKELY	0 PERSONS	6 PERSONS
EVACUATION	NUMBER OF PEOPLE	AREA
	67	2 SQ. MILES

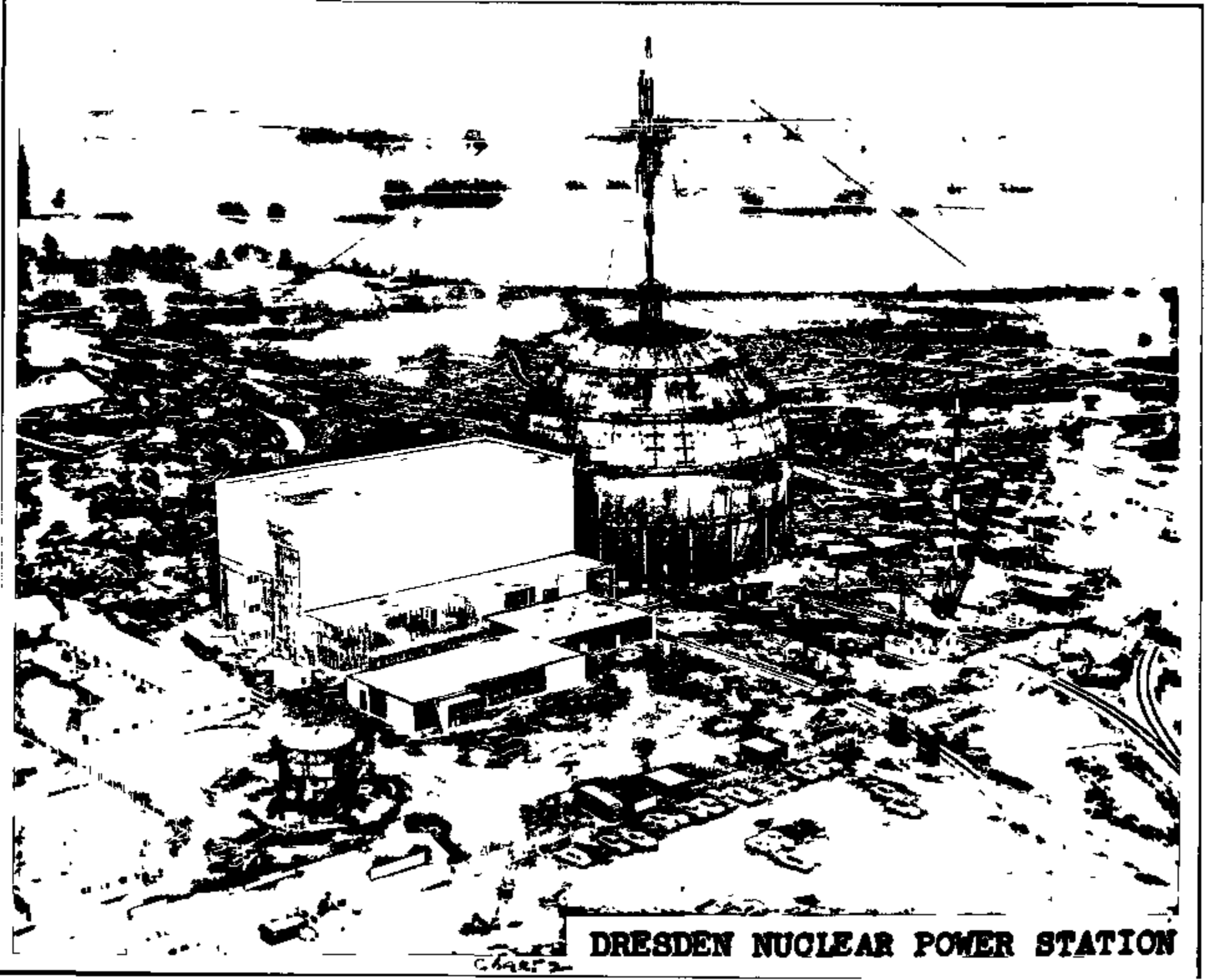
TABLE II

THE 50% RELEASE CASE

(ASSUMING A 2,000 FOOT EXCLUSION DISTANCE)

	MINIMUM (FAVORABLE CONDITIONS)	CONDITIONS OVER 3/4 OF TIME	MAXIMUM (WORST CONDITIONS)
PERSONAL INJURY		Less than:	
A. LETHAL EXPOSURE	0 PERSONS	10 PERSONS	3,400 PERSONS
A. LETHAL DISTANCE	0 MILES	1 MILE	15.5 MILES
B. INJURY LIKELY	0 PERSONS	100 PERSONS	42,500 PERSONS
B. INJURY DISTANCE	0 MILES	2.5 MILES	44 MILES

CHART 2



DRESDEN NUCLEAR POWER STATION

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DRAFT
CKBeck:vl
5/18/65

The Honorable Chet Holifield
Chairman, Joint Committee on Atomic Energy
Congress of the United States

Dear Mr. Holifield:

During the Joint Committee hearings last year, you suggested that concurrently with the anticipated consideration of extending the Price-Anderson Indemnity Act, re-examination should perhaps be made of the 1957 report prepared by the Atomic Energy Commission with the assistance of Brookhaven National Laboratory on "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants". This report is identified as WASH 740.

Pursuant to your suggestion, we asked the Brookhaven National Laboratory to re-evaluate its earlier study in light of today's technical information and current upward trends in power reactor sizes. It is the purpose of this letter to summarize the results of the laboratory re-evaluation, and to interpret the significance of these findings in the context of the 1957 calculations and in the context of present engineering practices and licensing procedures.

A Steering Committee*, composed largely of members from the Commission staff and from contractor organizations, met with Brookhaven at the outset to plan the re-study and again at intervals thereafter to discuss the course of the work.

---

* Dr. Frank Gifford, Dir., Dept. of Turbulence & Diffusion, U.S.Weather Bureau, Oak Ridge.
Dr. David Okrent, Senior Physicist, Director's Office, Argonne National Laboratory
Mr. James E. McLaughlin, NYOO
Dr. W. D. Claus, Special Assistant to Director, B&M, AEC
Dr. R. L. Doan, Director, RL, AEC
Mr. W. J. McCool, Asst. Dir. for Reactor Safety, OS, AEC
Mr. U.M. Staebler, Associate Dir. for Power Applications, RD&T, AEC
Dr. F. Western, Director, SS, AEC
Dr. Clifford K. Beck, Dep. Dir. of Regulation, AEC (Chairman of the Committee)

Transmitted to Commissioners by memo 5-21-65

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Brookhaven's letter report to us on the summary of results from their re-examination is attached as Part II.

In the original study in 1957, Brookhaven concentrated most of its efforts on calculating the theoretical consequences of major hypothetical accidents without attempting to assess the details of possible malfunctions that might lead to such accidents or the probability of occurrence of such events. The same pattern was followed in the present re-study. On the other hand, it is clearly the case that theoretical calculations of consequences that might derive from hypothetical accidents do not alone convey a true perspective of the risk or hazard to public health and safety which arises from operation of nuclear reactors. As a necessary accompaniment, there must also be a recognition of the likelihood or more accurately of the improbability of such events. Therefore, in Part I of the attachment there is presented a brief discussion of factors which help to understand Brookhaven's theoretical calculations in relation to the extremely low probability of such events.

In its summary of the present re-study of theoretical consequences of hypothetical accidents, Brookhaven reminds us that reactors of today are much larger than reactors in prospect in 1957, that their fuel cycles are longer and hence, that fission product inventories are larger. Then, to no one's surprise, they reach the basic conclusion for these reactors that, assuming the same kind of hypothetical accidents as those considered in the 1957 study, the theoretically calculated damages would not be less, and under some circumstances would be substantially more than the consequences reported in the earlier study. Brookhaven further comments significantly that "considerable improvement in numbers of engineering safeguards and their reliability has occurred in the intervening period. The effect of this trend can only be to reduce even further the probability of reactor accidents and to lower the consequences of any failures."

This latter observation is consistent with the discussion in Part I on those factors which increase our confidence in the consistently held opinion among experts that the likelihood of accidents of major dimensions in power reactors is extremely low. In the period since the 1957 study there has been substantial increase in knowledge about reactors and their operational characteristics, many improvements in reactor safeguards have been made, and our added years of operational experience with power reactors has been unblemished by any reactor accident that caused substantial damage to the facility or injury to any person. No accident in these facilities has caused interference with the public in any way.

In presenting this brief summary to you, we want to emphasize that there is with nuclear reactors, as in many other enterprises, a vast difference between potential hazards and the practical likelihood (or experience) of realized damages. It is the potential hazard inherent to nuclear reactors from their accumulations of fission product inventories and the conceivable consequences of accidental releases of such inventories that Brookhaven has considered. Assessment of the practical likelihood of realized damages must also take into account the multiple system of defenses provided to prevent the potential hazards.

Such defenses against the potential hazards of nuclear reactors permeate the basic design of every component and system, they are evident in the successive barriers against fission product release built into the accident-prevention and the consequences-limiting safeguards mentioned above; and the several independent safety reviews of every facility by reactor experts give assurance that no possible route to danger has been left unprotected. A firm basis has been laid for the belief that the likelihood of major accidents of the type considered by Brookhaven is extremely low.

Nevertheless, as pointed out in Part I of the attached memorandum, the possibility that such accidents might occur cannot be excluded. or rather, the nature of this problem is such that a quantitative measure cannot be objectively established of the probability (or improbability) of such conceivable accidents. It is probably impossible to develop a positive means to demonstrate that no such event could ever happen.

Despite this inherent element of uncertainty, the Commission, in transmitting this report, affirms its basic conviction that reactors with their multiple systems of protection do not present an undue hazard to the health and safety of the public.

We are encouraged by the present status of power reactors and the systems of safeguards which have been and are continuing to be developed against their well recognized potential hazards. But we are not, we cannot be, complacent. The re-study reported here of this old problem gives us renewed determination and gives to our research and regulatory surveillance efforts renewed vigor toward the objective that these potentials for public harm shall never be realized.

Sincerely yours,

Attachments:
As stated

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PART I

FACTORS RELATING TO THE PROBABILITY OF CATASTROPHIC REACTOR ACCIDENTS

Clifford K. Beck
Deputy Director of Regulation
U.S. Atomic Energy Commission
Washington, D.C.

DRAFT --- 5/18/65

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Potential Hazard vs. the Practical Likelihood of Realized Damage:

In no other endeavor undertaken by humans has the potential hazard been so great in proportion to the exceedingly small magnitude of actual realized damages as in the operation of nuclear reactors. The danger or risk to people consists of some combination, such as the product, of the inherent potential for hazard and the probability of that hazard being realized. Neither the magnitude of the potential hazard nor the size of the probability of its realization alone can furnish an accurate measure of the risk to people.

It is the first of these, the potential hazard inherent in nuclear reactors, that was considered by Brookhaven in their calculations reported in WASH 740, in 1957, "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants". A brief description of procedures and a summary of results of their re-examination in 1965 of theoretical consequences of hypothetical accidents for reactors now in prospect are reported in Part II of this document.

BNL Approach to Consequence Calculations:

Brookhaven's 1965 re-evaluation of the 1957 report is based on a "maximum consequence" accident postulated to occur as a result of a total loss of coolant coupled with a concurrent inability to prevent the core from melting and the failure of various successive barriers which would prevent release of radioactivity to the environment.

The 1965 study differs from the 1957 study in that it has been carried out on the basis of a water-cooled reactor of 3200 thermal megawatts (1000 mw) and a fuel irradiation period of 1000 days, such as are now being contemplated. The 1957 study was based on a water-cooled reactor of 500 thermal megawatts and a fuel reloading cycle of 180 days, which was in prospect at that time.

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Here is encountered the most baffling and insoluble enigma existing in our technology: it is in principle easy and straightforward to calculate potential damages that might be realized under such postulated accident conditions; there is not even in principle an objective and quantitative method of calculating probability or improbability of accidents or the likelihood that potential hazards will or will not be realized.

In the 1957 discussion of probability of catastrophic accidents, it was stated: "No one knows now or will ever know the exact magnitude of this low probability of a publicly hazardous reactor accident." It was then suggested that some estimation of this probability might be obtained from:

a. established or assigned numerical values to all separate factors which would either prevent or cause such an accident, then try to calculate the composite resultant probability of catastrophic accident;

b. operation of enough reactors for a sufficient length of time;

c. obtaining the best judgment and judicious opinions of knowledgeable experts.

None of these led to satisfactory estimates in 1957. Some further consideration is given below on items (a) and (b).

As to the opinion of experts, it was concluded that, "whether numerically expressed or not, there was no disagreement in the opinion that the probability of major reactor accidents is exceedingly low." No formal sampling of expert opinion has been made recently, but all available indications are that the opinion expressed in 1957 is even more firmly held today than it was in 1957.

<u>The Probabilistic Approach to Estimating the Likelihood of Reactor Accidents:</u>

The Planning Research Corporation, under contract to the Commission, has attempted to establish a "Probabilistic Method" of calculating the likelihood of accidents having damaging consequences to the public.* The first basic premise on

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*An investigation directed toward the principle objective of establishing a probabilistic reactor safety analysis methodology applicable to light water-cooled & moderated power reactor system, Contract AT(04-3)-570, Planning Research Corp., Los Angeles, Calif.

which this study rests is, that a sequence of events must occur for damage to the public to be realized.

For example, one of a finite number of combinations of failures, malfunctions, misoperation, etc. must occur before reactor fuel is damaged and fission products are released from the fuel; additional abnormalities and safeguard failures must occur to release the radioactivity from the primary system, still more failures must occur before radioactivity is released to the environment, and even then only under certain weather conditions, would prevailing dispersive conditions lead to hazardous contamination or exposure of people.

The second premise is that a probability value can be established for each of the sequential failures or abnormalities requisite to realization of public damage.

The final premise is that having determined the various possible sequences of events prerequisite to public damage and established probability values for each event in the different sequences, it will then be possible to calculate the over-all probability of each accident or of each magnitude of public damage.

In principle, there appears to be an appealing logic to these premises but there are many obvious questions about the practical means available for "establishing" values for the event-links in the sequence chain. There are still more questions about our present ability (a) to identify or to anticipate all the significant accident sequence-chains, and (b) to develop appropriate weighting formulas for the calculations. Thus, a major element of uncertainty must be recognized in any tentative results from present applications of this method, though the effort itself is a valuable one and hopefully more definitive applications can be developed in the future.

The Planning Research Corporation has developed a complete formulation for this "probabilistic method" of calculating reactor accidents and has made tentative calculations of catastrophic accidents for present water type reactors. They conclude:

".... The quasi-probabilities for the occurrence of catastrophic nuclear reactor accidents obtained from the reliability models are of the order of 10^{-k} per year where k lies between 8 and 16. Considering all catastrophic accidents applicable to the reactor types studied, the estimated probability of a catastrophic accident for a given type is in the range of 5×10^{-11} to 7×10^{-8} per reactor year of operation."

That is, from this study, tentative estimates are that the probability of publicly hazardous accidents would be in the range from 1 event in 15 million/reactor year to 1 event in 2 trillion/reactor year.

Thus, from this method, it is obvious that we are not now able to assign firm quantitative measures to our belief about the "low probability" of major reactor accidents. But these figures, whatever their worth, do not contradict this belief.

Factors contributing to Confidence that the Probability of Reactor Accidents is very low:

There are several specific factors which support our belief in the increased safety of nuclear power plants and, at the same time, a few elements prevent our being able to say with finality that such accidents cannot occur.

Much Progress has been made in Reactor Safety Research:

The extensive reactor research programs, which currently are being supported to the extent of many millions of dollars annually for efforts directly committed to exploration of reactor safety, have revealed a greatly expanded understanding of basic reactor characteristics and behavior. In the following areas of reactor technology, our basic knowledge has been substantially increased over the past 8 years:

a. Through power excursion experiments (SPERT, BORAX, etc.) our knowledge of the dynamic behavior of reactors in reactivity transients has been greatly expanded.

b. Our knowledge of the inherent coefficients and parameters which lead to automatic termination of reactivity excursions in water reactors with low enrichment oxide cores has been clarified.

c. Through observation of many planned experiments and analysis of a few unplanned incidents, our knowledge of factors affecting potential reactions between water and various metals commonly used in core construction has been extended.

d. Through extensive small scale experiments, currently being expanded into experiments utilizing large scale equipment, the factors affecting potential fractional release of fission products from damaged reactor fuel have been considerably clarified.

e. Our understanding of radiation embrittlement of structural components of reactors has been extended somewhat and, more importantly, a massive long range program is beginning to yield a necessarily slow but valuable accumulation of data on this phenomenon. Factors affecting radiation embrittlement and its reversibility, are being systematically explored.

f. Energy releases in potential reactor accidents, the subsequent pressure and temperature transients, and design of structures to accommodate these have been extensively explored. For example, engineering experiments on scale models have demonstrated that with properly designed containment buildings, "upper limit" energy releases in fast reactors will not result in breach of the containment by ejection of top shield plug, or other potential missiles.

Much new knowledge about reactor behavior and reliability has also been gained by successful design and operation of successive "generations" of reactors, particularly of the water type, and by experimental variation of

components and operating modes in such reactors. Much credit must be given to reactor manufacturers and owners for the contribution to basic understanding of reactors and hence to reactor safety, which has emerged from this work:

a. The stability, controllability and predictability of behavior of certain types of reactors have been amply demonstrated.

b. The technology of controlling and predicting power distribution in reactors has been greatly advanced.

c. Knowledge of burn-out ratios, safe limits on heat fluxes and hence of factors affecting safety margins in thermal design have been clarified.

d. The technology of efficiency in fuel loading and fuel cycles, made possible by predictability in reactor characteristics and behavior, through flux flattening, power density control, burnable poisons, chemical shims, etc. has been greatly extended with concomitant implications for safety of reactors.

e. The technology of fuel element fabrication, quality control, etc. leading to extended life expectancy, has also given greater confidence in safety.

Substantial improvements have been made in Engineering Safeguards:

When the Commission's guide on selection of sites for power reactors was published in 1961, the principle was established that, if adequate and dependable engineering safeguards were included in the design of the reactor, there could be some reduction in distance requirements that otherwise would be necessary. A strong incentive was therefore placed on the reactor manufacturers and owners to develop improvement in the safeguard systems in order to gain approval of less isolated reactor sites. In consequence, many new and fruitful ideas on reactor safeguard systems have been generated, and great strides have been made in improving the effectiveness and reliability of such systems.

In safeguards directed toward prevention of reactor accidents, much general progress and some specific advancements since 1957 have been accomplished. For convenience, an "accident-prevention" safeguard might be defined as any device or system intended to prevent some mishap - breakdown of machinery, component failure or malfunction, operator error, etc. from escalation into, say, a significant release of fission products from the reactor fuel. Thus, to prevent an inadvertent reactivity insertion from causing a major power excursion and fuel meltdown, a system for rapid automatic rod insertion or reactor scram is provided. To prevent a loss of coolant from causing fuel overheating and meltdown, an emergency cooling system is provided.

In addition to these two examples, accident prevention safeguards would include:

a. A poison injection system to assist in emergency shutdown.

b. Independent source(s) of main station power, in case the primary source of power should fail.

c. A high pressure core spray system to cool the fuel, in case circulation line, etc. are inoperative.

Aside from specific accident-prevention safeguard systems, substantial progress in the prevention of accidents has also been made through development of more definitive standards, codes and specifications for the design and construction of reactor pressure vessels, the valves and piping of the primary system and other vital reactor components.

Perhaps even more obvious advancements have been made in safeguard systems directed toward minimizing the hazardous consequences of a major accident, should one occur. No substantial hazards to the public arise from

accidents in reactors unless significant portions of the radioactivity inventory should escape from the reactor facility into the environment. Therefore, in conformity with the definition of accidents given above, "consequences-limiting" safeguards would have as their central purpose the prevention of fission product release to the environment which might have accidentally escaped from the fuel.

Among such safeguards which have been developed are the following:

a. Improved sources of auxiliary power to keep essential equipment in operation in case of failure of normal and emergency power supplies.

b. Spray-washdown systems to quench steam, reduce air temperature, and wash down fission products in the containment in case of accidents.

c. Filter systems to collect fission products by internal recirculation of building atmosphere.

d. Containment buildings of high strength and low-leakage. A number of improved new containment concepts have been evolved, such as:

(1) Double-containment-with-pumpback, and extra shielding outside

(2) Dry well containment with vapor suppression

(3) Double vapor suppression

Such safeguard systems provide a basis for increased confidence in the level of protection against potential accidents in the operation of nuclear reactors.

Accumulated reactor operating experience has been greatly expanded and continues to be highly successful:

In any machinery as complex as a reactor facility it is inevitable that structural failures, instrument malfunctions, operators' errors and other mishaps will occur, despite the most careful design and rigid schedules of

maintenance and administrative control. Such has been the experience with reactor installations. Analysis of the reactor accident experience over a sufficient period of time would provide a basis, actually the only reliable one, for estimating the actual hazard to the public from such facilities.

Some pertinent facts and observations on the extent of our experience to date are presented here:

1. Reactor Accidents:

For the 250 odd reactors*, built over the 22 year period since the first reactor in 1942, and operated for a total of about 1500 reactor years, only some 20 reactor accidents have occurred of sufficient severity to cause injury to people or significant damage to the facility. No reactor accidents have occurred of sufficient magnitude to cause injury to members of the general public; only 3 workers have been killed, all in one accident; very few other workers have received overexposure to radiation.

Less than half dozen such accidents have occurred in the ____ large power, production or propulsion reactors.**

No damaging accidents have occurred in the ___ power reactors now in use for central station electric power generation.

Most of the damaging accidents by far have occurred in small experimental, developmental or research facilities, often purposely located at remote sites because of the flexible exploratory nature of their programs. Such was the case with the reactor in which 3 workers were killed.

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* This tabulation does not include critical facilities or accidental criticality events in fuel processing plants.

** Loss of the Thresher is an exceptional case, not included in this tabulation.

Statistical Significance of Accident Record:

What can one say about the probability of catastrophic, publicly hazardous reactor accidents from the experience to date? There have been no such accidents in approximately 250 reactors operated for a cumulative total of about 1500 reactor years.

For simplicity of calculation it is assumed that all the reactors are alike as to their probability of catastrophic accidents. Then, based on the further assumption that the occurrence of catastrophic reactor accidents follows a normal Poisson distribution, simple mathematics would show that the probability of catastrophic accidents, at the 95% level of confidence, is not more than 1/500 per reactor year of operation. The actual probability is certainly a great deal smaller, but the extent of reactor operation is simply insufficient to demonstrate this. For example, if there were 15,000 reactor years of experience without a catastrophic accident, the probability of such accidents, at the 95% confidence level, would be not more than 1/5000 per year.

Such an approach, of course, predicts that the probability of catastrophic accidents is not larger than 1 in 500 per reactor year, but it does not say how low this probability is --- which is the question of real importance.

Thus, it must be concluded that projection of our present reactor operating and accident experience cannot be used to predict an expected frequency of catastrophic reactor accidents.

Factors contributing to uncertainty:

Despite the basis of confidence in the safety of reactors provided by the many factors mentioned above, a few aspects of the present status of reactor technology prevent a firm and final conclusion that a major hazard to people from a reactor accident cannot occur. Among the more significant of these are:

a. Abnormal incidents: In all complex system of machinery, mishaps of many kinds occur. At every reactor facility a long list of operating abnormalities occur over a period of time, and a complete tabulation for all reactors would include thousands of such incidents. Breakdown of machinery, malfunction of instruments, deviations from established procedures and operators' errors would be among these incidents.

Such experiences give rise simultaneously to both reassurance and to uncertainty. Reassurance arises from realization that programs of safety surely must be large for this number of mishaps to have resulted in so few damaging events, mentioned above, and no events of magnitude to have caused damage to the public. On the other hand, totally unexpected abnormal situations do occur, and it is the case that relatively minor events in themselves, in combination with other abnormalities, can turn an insignificant situation into a major accident.

b. Serious Incipient Failures: There have been discovered in reactor systems a few incipient failures which, had complete failure occurred, would have resulted in more serious accidents than any thus far experienced. As examples, in three reactors two or three of the stud bolts on the head closure of the main pressure vessel or at crucial locations within the pressure vessel were badly cracked or broken. In another reactor two main control rod shafts

were found to be cracked from stress corrosion; in two reactors significant cracks were found in the piping of the main primary coolant system. Small, easily visualizable extensions of these situations could have led to serious accidents, though still by no means necessarily to major public hazards, for additional protective safeguards would still have remained.

c. Many of the more imaginative and extensive safeguard systems designed for power reactors have not yet been proven out by simulated tests or operating experience on the reactor facility of which they are a part.

d. Because of the high level of induced radioactivity, the interior surfaces of major components of the primary system cannot be inspected for evidence of incipient failures. These are the vital components where major ruptures would have most serious consequences.

e. No large power reactors have yet been carried through a major portion of their expected lifetime. It is not yet known what pattern of long term cumulative effects leading to failures will be revealed in the latter half of reactor life.

f. In a few areas our safety research programs are addressed to difficult technical problems on which sufficient information has not yet been established to give complete assurance that all necessary and appropriate precautions are being taken. Among these, the embrittlement of structural components by irradiation and the possibility of chemical reactions between water and certain metals frequently used in reactor structures are examples.

g. Reactor technology is still in a relatively early stage of development and it is not yet certain that all possibilities of unsafe behavior have been identified and appropriately safeguarded. As an example, there occurred in the SL-1 accident a completely unexpected water-hammer phenomenon which projected the massive pressure vessel completely out of its normal location and sheared all connections to auxiliary apparatus.

SUMMARY:

From all the information available about reactors and their behavior and from the cumulated reactor experience to the present time, there is a basis for confident belief that the likelihood of occurrence of reactor accidents of sufficient severity to endanger the public is exceedingly low, even lower than would have been estimated at the time of the first report on Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants.

On the other hand, the possibility that such accidents might occur cannot be excluded and there has been accumulated some evidence that a few failures may have almost occurred which could have resulted in more serious accidents than any which have thus far been experienced. However, even had these incipient failures resulted in major breaches of the primary cooling systems, there still would have remained several safeguard barriers against the release of radioactivity in significant amounts to the environment.

Thus, all the evidence available to the present time indicates that the probability of accidents in the publicly hazardous category of those analyzed in the summary report from Brookhaven attached hereto have exceedingly low likelihood of occurrence, though nuclear technology is now and probably will remain for a long time in the uncomfortable position where neither the impossibility nor the probability of these conceivable occurrences can be positively demonstrated or quantitatively measured.

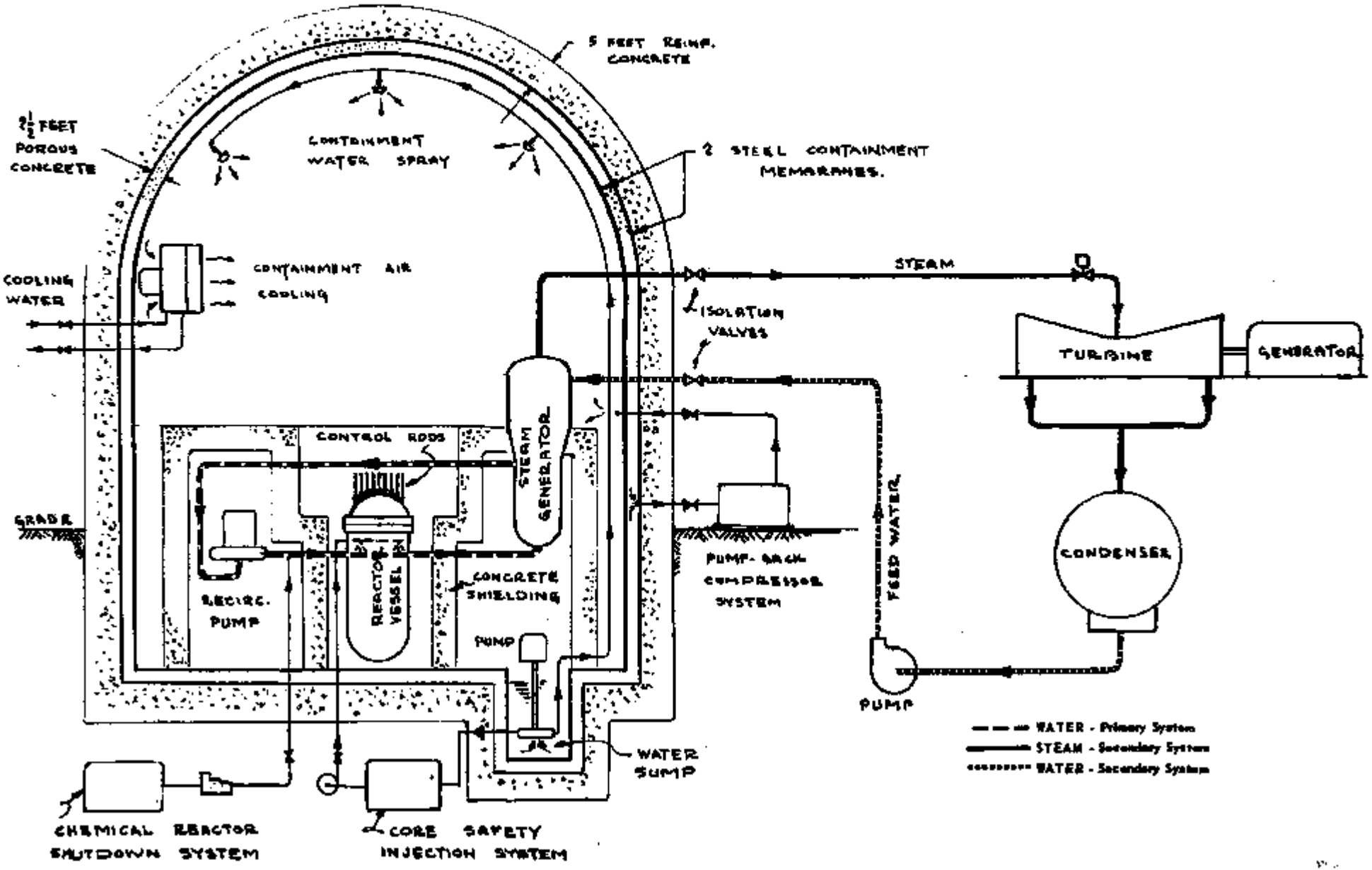
PRESSURIZED WATER REACTOR

IN

DUAL BARRIER CONTAINMENT

FIGURE - 1

BASED ON MALIBU REACTOR FACILITY



PRESSURIZED WATER REACTOR IN DUAL BARRIER CONTAINMENT

BASED ON MALIBU REACTOR FACILITY

In this pressurized water reactor, heat is absorbed by water in the primary coolant system flowing through the reactor vessel and is transferred to the secondary coolant system in a steam generator where steam is produced and delivered to a turbine located outside of the containment structure.

REACTOR VESSEL - Serves to contain the nuclear core which heats the recirculated water in the primary coolant system.

Design Pressure	- 2485 pounds per square inch.
Design Temperature	- 650° F.
Thickness and Diameter	- 10-1/2 inches by 14-1/2 feet.

CONTAINMENT VESSEL - Serves to prevent the leakage (within limits) of radioactivity released to the containment atmosphere in the event of an accidental meltdown of the nuclear core.

Leakage from containment vessel limited by 2 steel membranes separated by 2-1/2 feet porous concrete zone, maintained at a pressure less than atmospheric by pump-back compressor system.

Outer 5 ft. concrete shell serves to withstand internal pressure and also as a biological shielding for personnel protection.

Design Pressure	- 40 pounds per square inch.
Allowable Leakage Rate	- 0.1% of contained volume per day.
Thickness of Membranes	- 3/8 inch.
Diameter of Vessel	- 135 feet.

CONTAINMENT WATER SPRAY - Removes iodine in containment atmosphere by washing action, and condenses steam released by rupture of main coolant system.

CONTAINMENT AIR COOLING - Maintains air temperature below 120°F. during normal operation, and cools and condenses vapors in containment atmosphere to prevent overpressure due to decay heat generation.

CHEMICAL REACTOR SHUTDOWN SYSTEM - Provides for injection of boric acid solution (neutron absorber) into the reactor vessel to hold the nuclear core sub-critical in the event neutron absorbing control rods fail to function.

CORE SAFETY INJECTION SYSTEM - Provides for injection of water into the reactor vessel in the event of a large primary coolant system rupture in order to maintain core cooling and prevent fuel element meltdown due to decay heat generation.

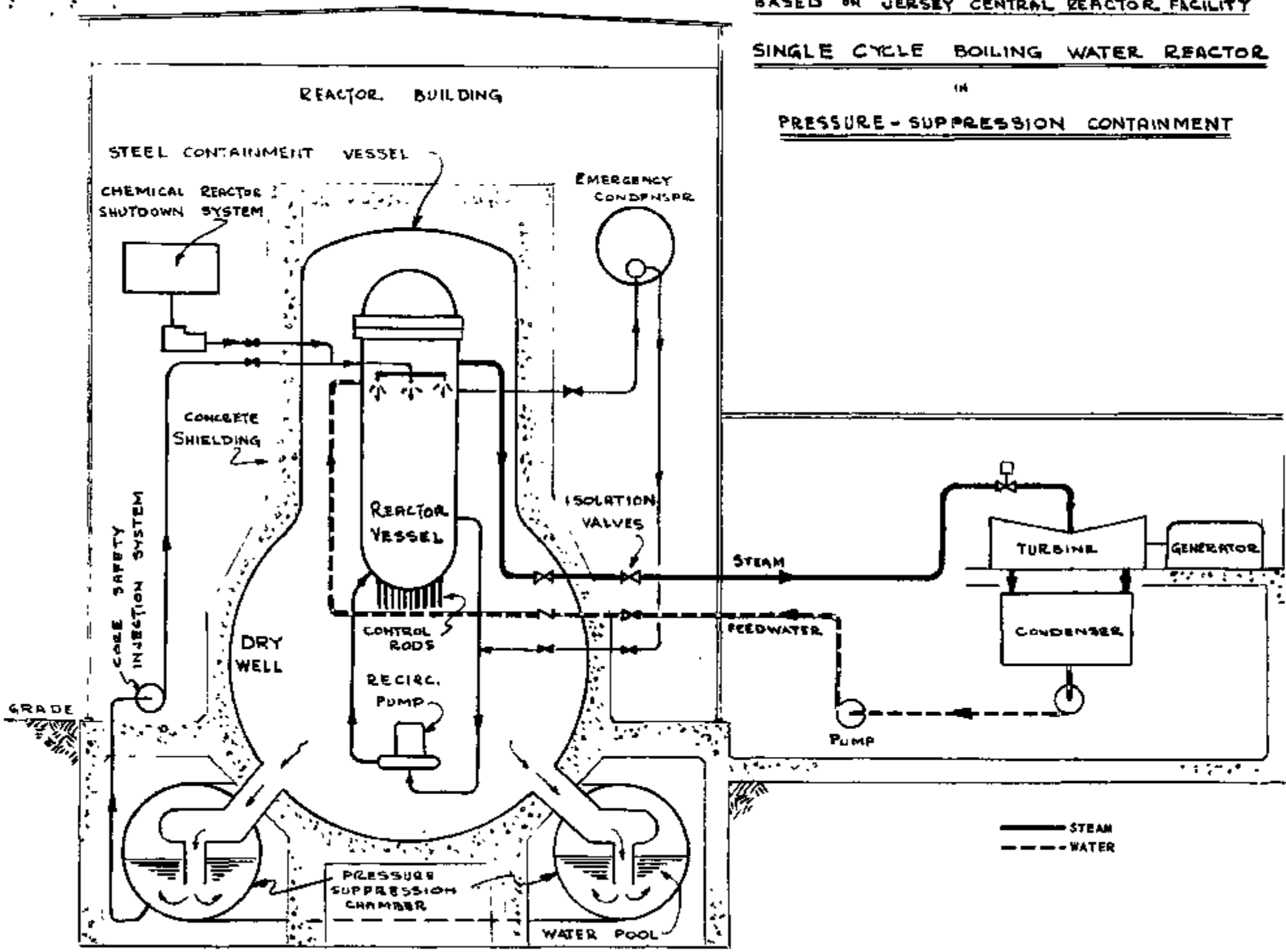
FIGURE - 2

BASED ON JERSEY CENTRAL REACTOR FACILITY

SINGLE CYCLE BOILING WATER REACTOR

IN

PRESSURE - SUPPRESSION CONTAINMENT



SINGLE CYCLE BOILING WATER REACTOR IN PRESSURE-SUPPRESSION CONTAINMENT BASED ON JERSEY CENTRAL REACTOR FACILITY

In this single cycle boiling water reactor, the water in the primary coolant system flowing through the reactor core is converted directly into steam which is in turn delivered to the turbine located outside of the containment vessel.

REACTOR VESSEL - Serves to contain the nuclear core which heats the recirculated water in the primary coolant system to the boiling point.

Design Pressure	- 1250 pounds per square inch.
Design Temperature	- 575° F.
Thickness and Diameter	- 7-3/8 inches by 17-2/3 feet.

CONTAINMENT VESSEL - Serves to prevent the leakage (within limits) of radioactivity released to the containment atmosphere in the event of an accidental meltdown of the nuclear core. The flashing steam and water in the containment atmosphere (dry well) resulting from a rupture of the primary coolant system is directed into a pool of water (pressure-suppression chamber). Upon condensation, a rapid reduction in the overpressure in the containment results.

Design Pressure	- 62 psig (dry well) 35 psig (pressure-suppression)
Allowable Leakage Rate	- 0.5% contained volume per day.

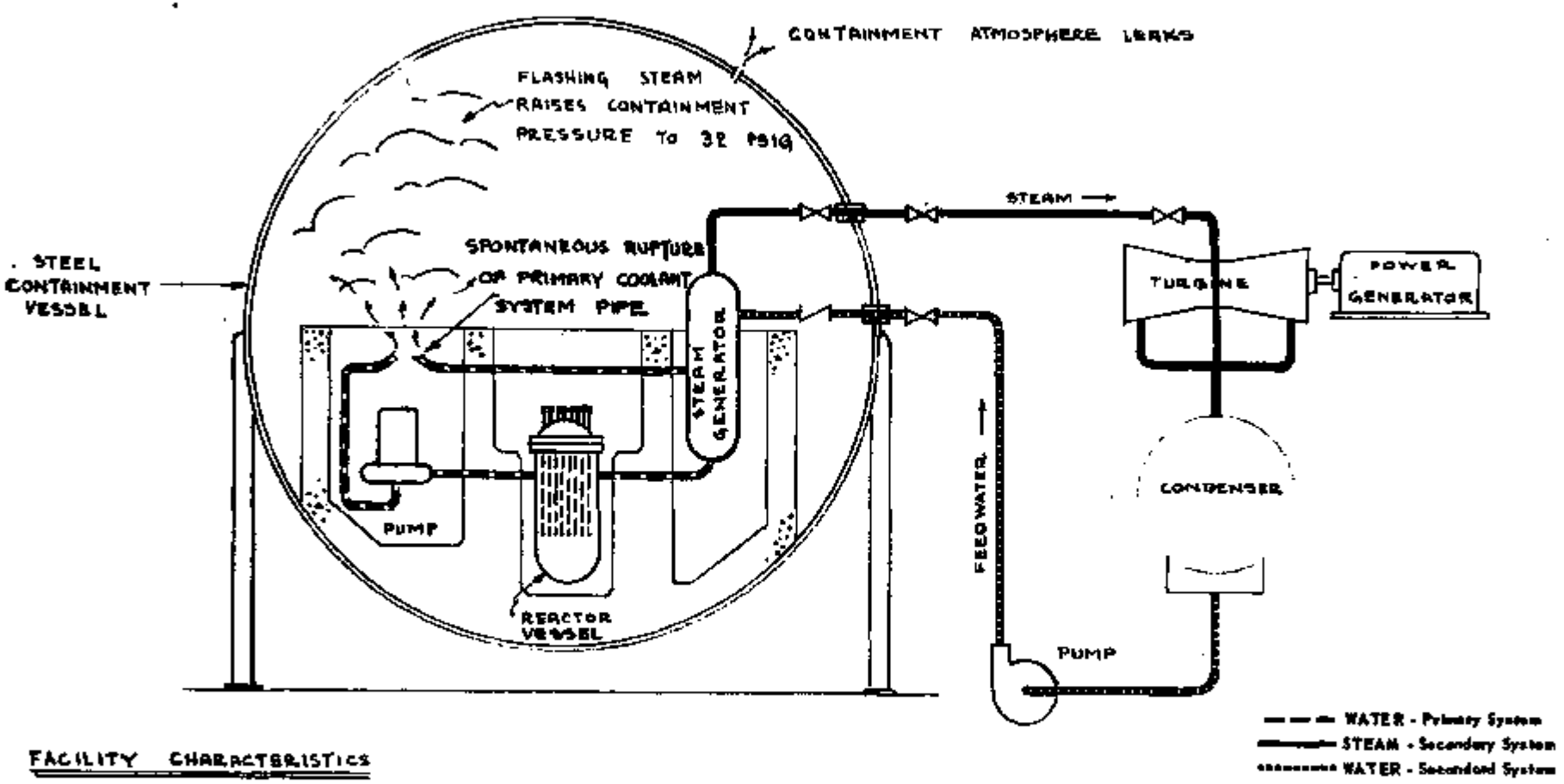
EMERGENCY CONDENSER SYSTEM - Provides a cooling system to dissipate reactor decay heat following reactor scram requiring closure of the isolation valves in the primary coolant system.

CHEMICAL REACTOR SHUTDOWN SYSTEM - Provides for injection of boric acid solution (neutron absorber) into the reactor vessel to hold the nuclear core sub-critical in the event neutron absorbing control rods fail to function.

CORE SAFETY INJECTION SYSTEM - Provides for injection of water into the reactor vessel in the event of a large primary coolant system rupture in order to maintain core cooling and prevent fuel element meltdown due to decay heat generation.

FIGURE - 3

ACCIDENT CONDITIONS POSTULATED FOR PRESSURIZED WATER REACTOR IN SINGLE BARRIER CONTAINMENT VESSEL



FACILITY CHARACTERISTICS

REACTOR

TYPE - PRESSURIZED WATER
POWER LEVEL - 600 THERMAL MEGAWATTS
PRIMARY SYSTEM PRESSURE - 2000 PSIG
PRIMARY SYSTEM TEMPERATURE - 632° F

ENGINEERED SAFEGUARD

STEEL CONTAINMENT VESSEL
DESIGN PRESSURE - 32 PSIG
DIAMETER - 125 FEET
SHELL THICKNESS - 7/8 INCH
SPECIFIED LEAKAGE - 0.1 % PER DAY

MAXIMUM ACCIDENT ASSUMPTIONS AND CALCULATED POTENTIAL CONSEQUENCES

BROOKHAVEN REPORT ANALYSIS	REGULATORY STAFF ANALYSIS
1. Loss of coolant accident occurs and results in a 100% core meltdown and rupture of the containment vessel. 100% of noble gas, 100% of iodine and 1% of solid fission products escape from fuel to containment atmosphere.	1. Spontaneous double ended rupture of largest primary coolant system pipe connected to reactor vessel occurs inside of the containment vessel resulting in a loss-of-coolant accident. a. Pressure inside containment increases in 18 seconds from 0 psig to peak value of 32 psig. b. Coolant and moderator is lost from pressure vessel due to flashing, entrainment, and/or boiling. c. Reactor is shutdown but fuel cladding melts due to continued decay heat generation within fuel. d. 100% of noble gas, 50% of iodine and 1% of solid fission products are released from fuel to containment atmosphere.
	2. Pressure inside containment decreases from peak value of 32 psig to 15 psig in 20 hours due to condensing of steam on containment walls, and heat radiation and conduction from containment vessel to atmosphere.
	3. 50% of the iodine released from fuel is plated out on containment walls.
4. 100% of the noble gas, 100% of iodine and 1% of solid fission products, initially in fuel, leak from containment vessel as a puff type cloud release.	4. 100% of noble gas, 25% (50% of 50%) of iodine and 1% of solid fission products initially in fuel leak from containment vessel at a rate of 0.1% per day.
5. Doses at 1/2 mile from passing cloud are 2,700 rad whole body and 4,500,000 rad to the thyroid.	5. Doses at 1/2 mile in 2 hours are 1.7 rad whole body and 290 rad to the thyroid.
6. Doses from passing cloud at 7.5 miles are 240 rad whole body and 67,000 rad to the thyroid.	6. a. Doses at 7.5 miles in 24 hours are 0.3 rad whole body and 40 rad to the thyroid. b. Doses at 7.5 miles in 30 days are 0.6 rad whole body and 280 rad to the thyroid.

METHODS TO PREVENT UNCONTROLLED RELEASES OF FISSION PRODUCTS FROM REACTOR FACILITIES

1. Design and fabrication of fuel elements and their cladding so as to maintain cladding integrity during normal and abnormal operations of the reactor.

2. Design and provision of reliable instrumentation system to measure operating parameters of the reactor.

3. Design and provision of reliable reactor safety system to cause initiation of reactor shutdown in the event of abnormal operating conditions.

4. Design and provision of reliable control rod drive system to shutdown the reactor in the event of abnormal operating conditions.

5. Design and provision of emergency chemical reactor shutdown system to hold the nuclear core subcritical in the event control rods fail to function.

6. Design and fabrication of primary coolant system to maintain its integrity under normal and abnormal operating conditions.

7. Design and provision of coolant purification system to maintain proper coolant chemistry in order to avoid corrosion of primary system materials.

8. Design and provision of pressure relief system to maintain coolant pressure in the primary system within allowable limits.

9. Design and provision of emergency condenser to provide a cooling system to dissipate reactor decay heat following reactor scram requiring closure of isolation valves in the primary coolant system.

10. Provision of well-trained reactor operating staff with competent technical support staff.

11. Delineation of clear-cut lines of authority with a clear definition of the responsibility of individuals for safety of reactor operations.

12. Use of detailed written procedures covering all phases of reactor operation.

13. Provision for periodic tests of performance of reactor components and systems.

14. Provision for timely preventive and corrective maintenance of reactor components and systems.

METHODS TO REDUCE CONSEQUENCES OF UNCONTROLLED RELEASES OF FISSION PRODUCTS FROM REACTOR FACILITIES

1. Design and provision of core safety injection system for injection of water into the reactor vessel in the event of a large primary coolant system rupture in order to maintain core cooling and reduce amount of fuel element meltdown due to decay heat generation.

2. Design and provision of containment spray system to reduce containment pressure following an accidental release of steam from a rupture in the primary coolant system. The scrubbing action of the water spray also removes from the containment atmosphere some of the iodine vapors and solid radioactive contaminants which may be released from a fuel element meltdown.

3. Design and provision of containment air cooling system to cool and condense steam in containment atmosphere resulting from a rupture in primary coolant system rupture in order to prevent containment overpressure due to decay heat generation.

4. Design and provision of containment air filters to remove radioactive materials, particularly iodines, released to the containment atmosphere in the event of fuel element meltdown.

5. Design and provision of containment vessel to prevent the leakage (within limits) of radioactivity released to the containment atmosphere in the event of fuel element meltdown.

6. Design and provision of pump back compressor system to maintain a negative pressure between double containment shells and to pump any leakage from negative pressure zone into the inner containment shell.

7. Design and provision of emergency power system to provide necessary power in the event of loss of normal power supply.

CHART 6

SUMMARY OF USUAL PROCESSING TIME FOR CONSTRUCTION PERMIT FOR "CONVENTIONAL" WATER POWER REACTORS (JERSEY CENTRAL USED AS EXAMPLE)

Date		Item	Months Elapsed
3/30/64		Application for construction permit received.	0
5/1/64	Staff & ACRS Subcommittee Review (4-1/2 mos.)	ACRS Subcommittee and Regulatory Staff toured reactor site.	1
6/4/64		Regulatory Staff Request for additional information.	2
7/2/64		Amendment 2 submitted company in response to request for additional information	3
8/7/64		ACRS Subcommittee meeting.	4
8/13/64		Regulatory Staff Safety Analysis forwarded to ACRS.	4-1/2
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8/28/64		ACRS report to Chairman, AEC.	5
9/9/64	Staff Anal. (12 days)	Notice of public hearing published.	5
9/28/64	Board-Hearing (2 mos.)	Final Staff Safety Analysis made available to Hearing Board and to public.	6
10/14-16/64		Public Hearing, Tom's River, New Jersey.	6
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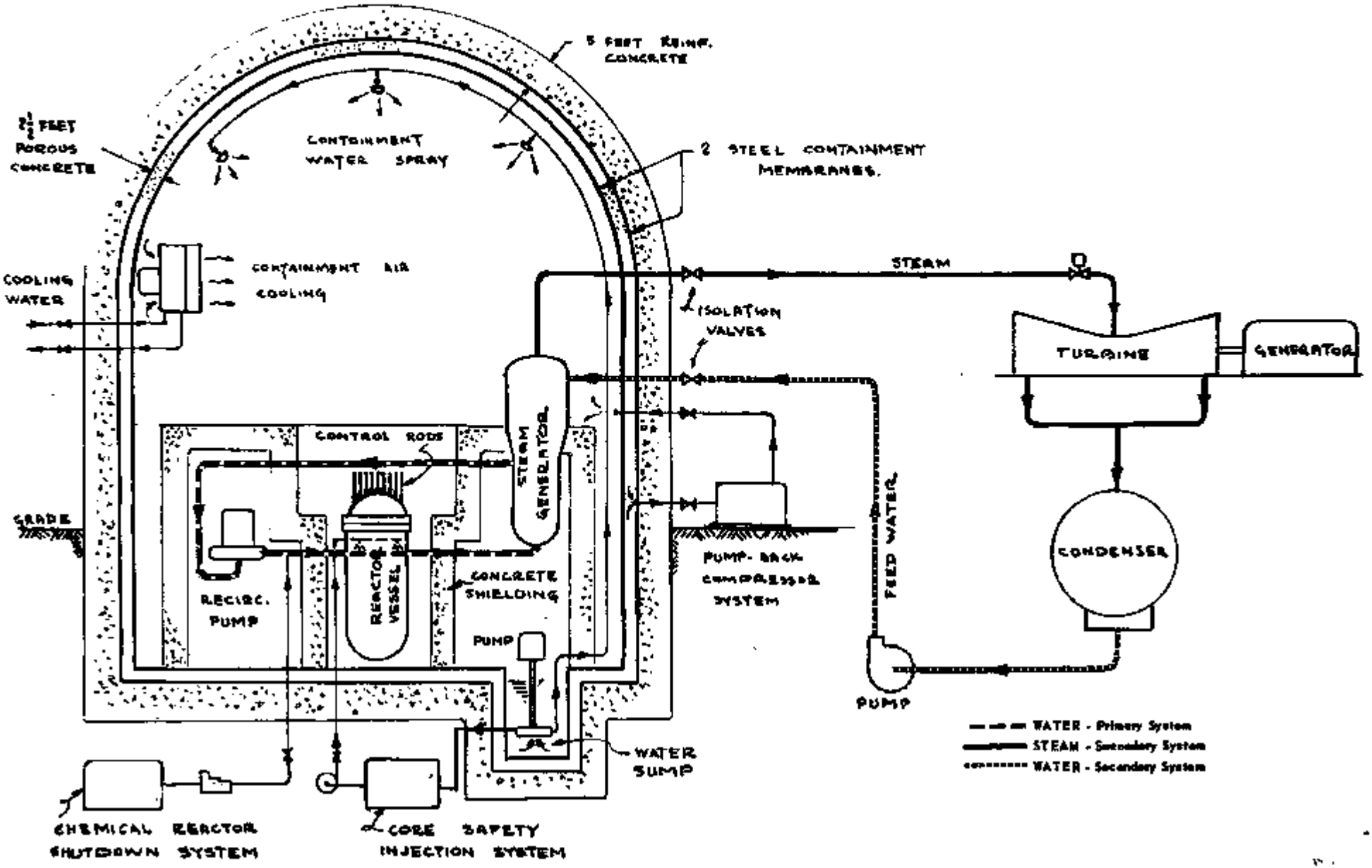
PRESSURIZED WATER REACTOR

IN

DUAL BARRIER CONTAINMENT

FIGURE - 1

BASED ON MALIBU REACTOR FACILITY



PRESSURIZED WATER REACTOR IN DUAL BARRIER CONTAINMENT

BASED ON MALIBU REACTOR FACILITY

In this pressurized water reactor, heat is absorbed by water in the primary coolant system flowing through the reactor vessel and is transferred to the secondary coolant system in a steam generator where steam is produced and delivered to a turbine located outside of the containment structure.

REACTOR VESSEL - Serves to contain the nuclear core which heats the recirculated water in the primary coolant system.

Design Pressure	- 2485 pounds per square inch.
Design Temperature	- 650° F.
Thickness and Diameter	- 10-1/2 inches by 14-1/2 feet.

CONTAINMENT VESSEL - Serves to prevent the leakage (within limits) of radioactivity released to the containment atmosphere in the event of an accidental meltdown of the nuclear core.

Leakage from containment vessel limited by 2 steel membranes separated by 2-1/2 feet porous concrete zone, maintained at a pressure less than atmospheric by pump-back compressor system.

Outer 5 ft. concrete shell serves to withstand internal pressure and also as a biological shielding for personnel protection.

Design Pressure	- 40 pounds per square inch.
Allowable Leakage Rate	- 0.1% of contained volume per day.
Thickness of Membranes	- 3/8 inch.
Diameter of Vessel	- 135 feet.

CONTAINMENT WATER SPRAY - Removes iodine in containment atmosphere by washing action, and condenses steam released by rupture of main coolant system.

CONTAINMENT AIR COOLING - Maintains air temperature below 120°F. during normal operation, and cools and condenses vapors in containment atmosphere to prevent overpressure due to decay heat generation.

CHEMICAL REACTOR SHUTDOWN SYSTEM - Provides for injection of boric acid solution (neutron absorber) into the reactor vessel to hold the nuclear core sub-critical in the event neutron absorbing control rods fail to function.

CORE SAFETY INJECTION SYSTEM - Provides for injection of water into the reactor vessel in the event of a large primary coolant system rupture in order to maintain core cooling and prevent fuel element meltdown due to decay heat generation.

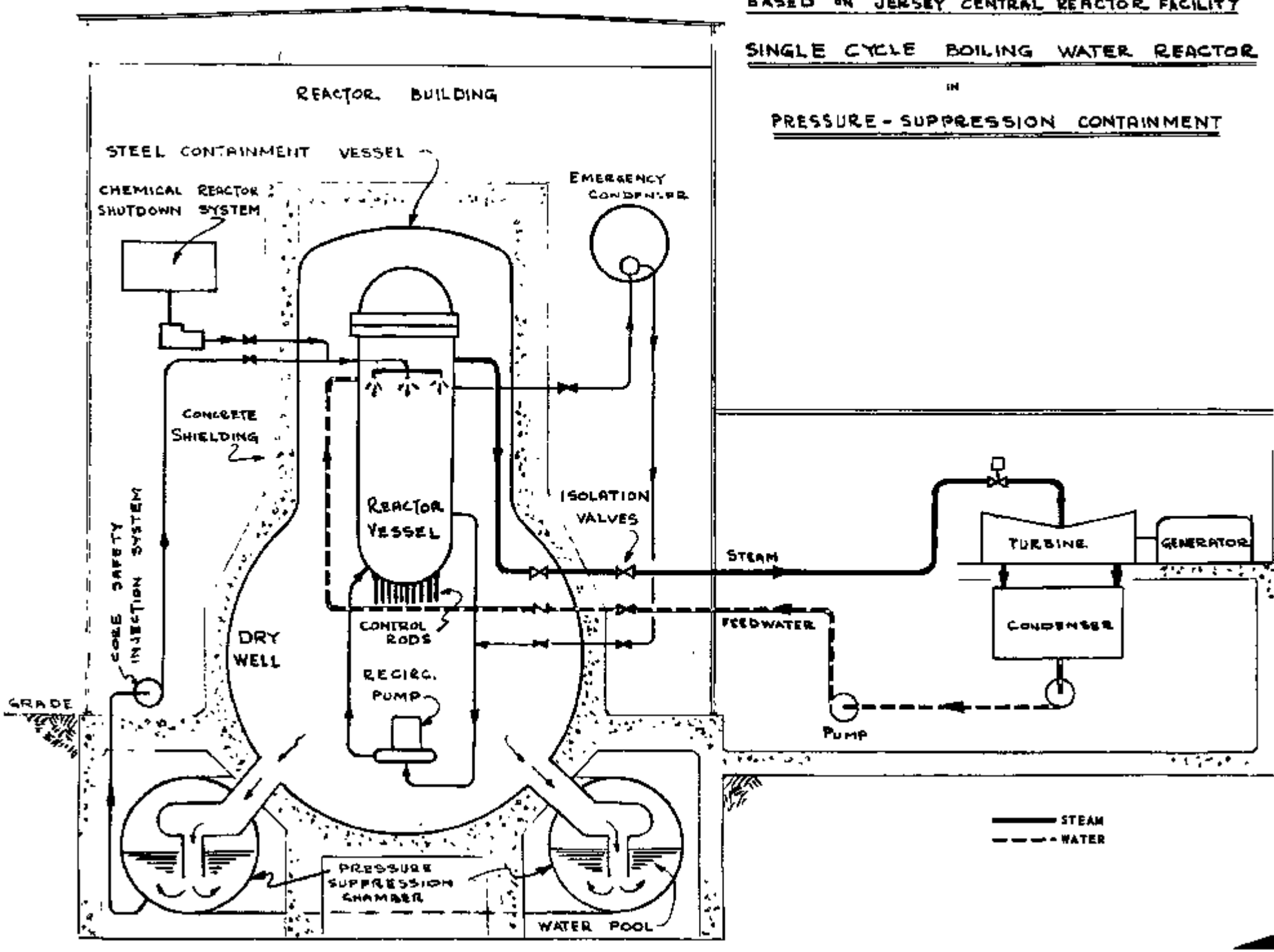
FIGURE - 2

BASED ON JERSEY CENTRAL REACTOR FACILITY

SINGLE CYCLE BOILING WATER REACTOR

IN

PRESSURE - SUPPRESSION CONTAINMENT



SINGLE CYCLE BOILING WATER REACTOR IN PRESSURE-SUPPRESSION CONTAINMENT BASED ON JERSEY CENTRAL REACTOR FACILITY

In this single cycle boiling water reactor, the water in the primary coolant system flowing through the reactor core is converted directly into steam which is in turn delivered to the turbine located outside of the containment vessel.

REACTOR VESSEL - Serves to contain the nuclear core which heats the recirculated water in the primary coolant system to the boiling point.

Design Pressure	- 1250 pounds per square inch.
Design Temperature	- 575° F.
Thickness and Diameter	- 7-3/8 inches by 17-2/3 feet.

CONTAINMENT VESSEL - Serves to prevent the leakage (within limits) of radioactivity released to the containment atmosphere in the event of an accidental meltdown of the nuclear core. The flashing steam and water in the containment atmosphere (dry well) resulting from a rupture of the primary coolant system is directed into a pool of water (pressure-suppression chamber). Upon condensation, a rapid reduction in the overpressure in the containment results.

Design Pressure	- 62 psig (dry well) 35 psig (pressure-suppression)
Allowable Leakage Rate	- 0.5% contained volume per day.

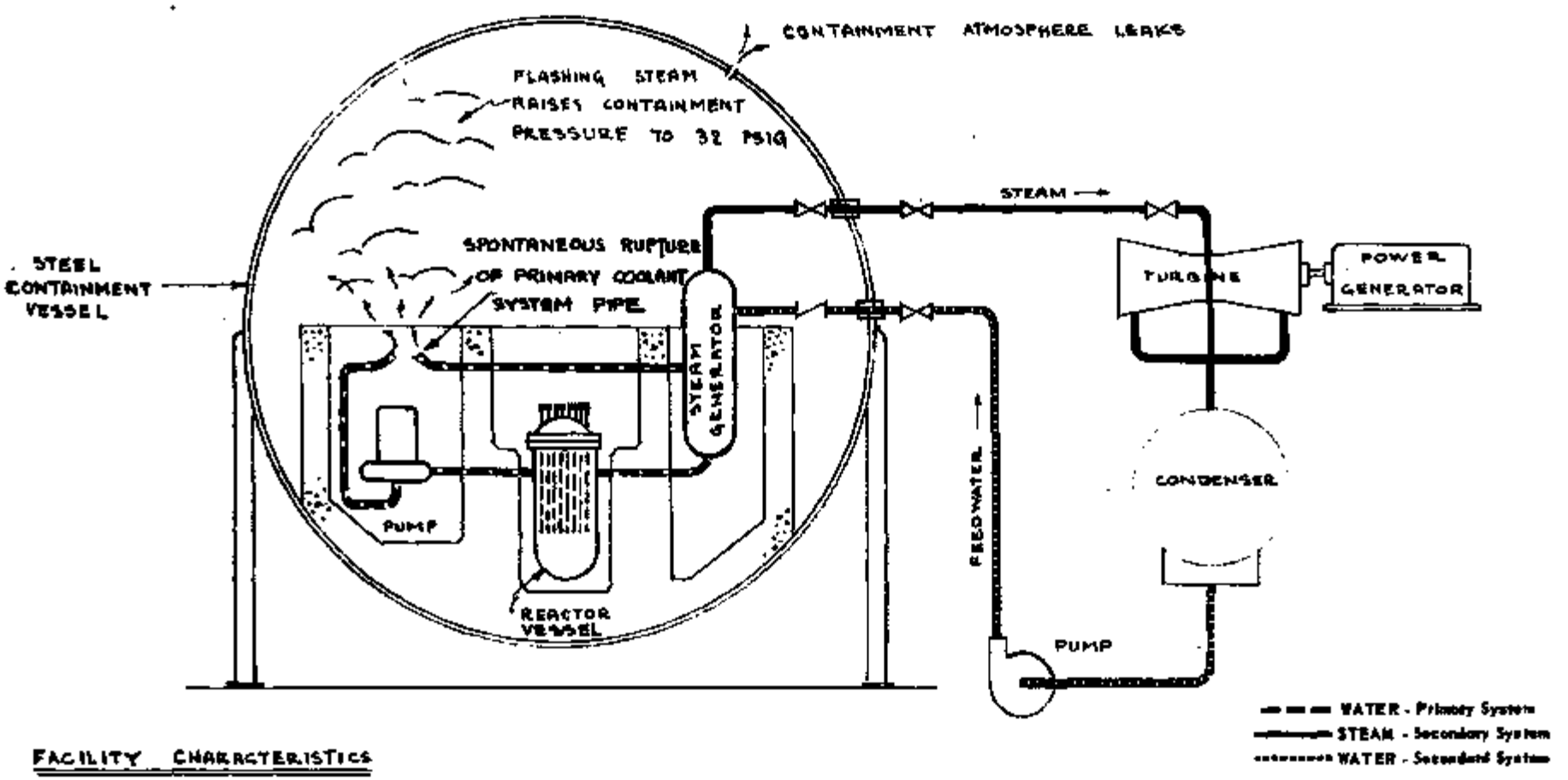
EMERGENCY CONDENSER SYSTEM - Provides a cooling system to dissipate reactor decay heat following reactor scram requiring closure of the isolation valves in the primary coolant system.

CHEMICAL REACTOR SHUTDOWN SYSTEM - Provides for injection of boric acid solution (neutron absorber) into the reactor vessel to hold the nuclear core sub-critical in the event neutron absorbing control rods fail to function.

CORE SAFETY INJECTION SYSTEM - Provides for injection of water into the reactor vessel in the event of a large primary coolant system rupture in order to maintain core cooling and prevent fuel element meltdown due to decay heat generation.

FIGURE - 3

ACCIDENT CONDITIONS POSTULATED FOR PRESSURIZED WATER REACTOR IN SINGLE BARRIER CONTAINMENT VESSEL



FACILITY CHARACTERISTICS

REACTOR

TYPE - PRESSURIZED WATER
POWER LEVEL - 600 THERMAL MEGAWATTS
PRIMARY SYSTEM PRESSURE - 2000 PSIG
PRIMARY SYSTEM TEMPERATURE - 532° F

ENGINEERED SAFEGUARD

STEEL CONTAINMENT VESSEL
DESIGN PRESSURE - 32 PSIG
DIAMETER - 125 FEET
SHELL THICKNESS - 3/8 INCH
SPECIFIED LEAKAGE - 0.1 % PER DAY

Chart 3

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BACKGROUND STATEMENT June 18, 1965

ON

REACTOR LICENSING PROCEDURES AND REACTOR SAFETY

PROCEDURES IN LICENSING NUCLEAR REACTORS

Protection of health and safety is the primary goal of the Atomic Energy Commission's regulatory program. Under this program, the licensed uses of radioactive materials and the construction and operation of licensed nuclear facilities, of which reactors are one type, are regulated by AEC. State and local officials are kept informed of AEC licensing actions taken in connection with a power reactor project.

The following is a general summary of the main steps in the licensing of a nuclear power reactor.

Selection of a Reactor Site -- Factors considered by the Commission in judging the safety of proposed sites for power reactors include dimensions and characteristics of the site under the operator's control; population density in the area surrounding the proposed site, and the uses which are made of this area, such as industrial, farming or residential; and the seismology, meteorology, geology and hydrology of the area. Other factors considered are the characteristics of the proposed reactor, including maximum power level, use of the facility, and the safeguards engineered into the plant either to prevent or to limit the consequences of accidents; and the extent to which the design of the reactor incorporates unique or unusual features which may have a significant bearing on the probability or consequences of an accident.

Before formally filing an application for construction and operation of a power reactor, a prospective applicant may discuss possible sites for the reactor with the Commission's regulatory staff and with the AEC Advisory Committee on Reactor Safeguards, a committee composed of scientists and engineers who are eminently qualified in the various fields related to reactor technology. The ACRS advises the Commission on safety aspects of reactors. In this way an applicant can receive additional guidance as to the acceptability of a site and the information which must be included in the application for a license to construct and operate a reactor.

CONTAINS NO DOE CLASSIFIED INFO
COORDINATED WITH GDO (Naval Reactors)
DOE EXEMPTION 5 U.S.C. 552(b) EO 12958
BY DOE DOCUMENT DOCUMENTS
AEC 7/7/99

Application for a Construction Permit -- Once a company has decided it wants to build a power reactor at a particular site, it must file an application with the Commission for a permit to begin construction. Such things as site exploration, site excavation, procurement or manufacture of components and construction of non-nuclear facilities may be done without an AEC construction permit.

Along with the application, the company files a preliminary safeguards report. This report sets out in as much detail as is then possible the technical aspects of the proposed reactor, as well as comprehensive data on the proposed site. The report discusses conceivable accidents which might occur and the safeguards which will be provided to prevent accidents, or if they should occur, to prevent overexposure of the public and employees to radiation.

The safeguards report is carefully studied by the regulatory staff. Copies are sent to the Commission's Advisory Committee on Reactor Safeguards, as well as to state and local officials, and are placed in AEC's Public Document Room. A public announcement of the receipt of the application is issued by AEC and a notice is published in the Federal Register. Copies of all correspondence and filings relating to the application are placed in the public records of the Commission.

The Atomic Energy Act includes provisions establishing the Advisory Committee on Reactor Safeguards as a statutory committee, requiring certain of its safety reports be made public, and requiring that public hearings be held on certain reactor license applications.

Review and Safety Evaluation -- The application is reviewed by technical experts of the Commission's regulatory staff. This staff is headed by the Director of Regulation, who reports directly to the Commission. The thorough technical review of the application includes consideration of all the radiation safety aspects of the proposed reactor, as well as the applicant's technical and financial qualifications. The regulatory staff supplements this study of the hazards summary report with conferences with conferences with the technical staff of the applicant, and may ask for further information if this is required in the safety determination which must be made.

The data submitted must provide the necessary information to permit evaluation of the adequacy of the proposed site for a reactor of the power level and type planned. Even though complete design details are usually not available at the time of the application for a construction permit, the data submitted must provide reasonable assurance that a reactor of the type and power level proposed can be constructed and operated at the selected site without endangering the health and safety of the public, including plant employees.

ACRS Safety Evaluation -- Before completing its review the regulatory staff analyzes the safety aspects of the proposed power reactor and prepares a summary of this analysis for the Advisory Committee on Reactor Safeguards, which has already received the applicant's preliminary safeguards report. Particular problems which may exist are identified for consideration by ACRS. Representatives of the applicant and members of the regulatory staff usually meet with the ACRS to deal with questions that arise during the Committee's review of the reactor. When it has reached a conclusion as to safety aspects of the proposed reactor, the ACRS reports its views to the Commission. This report is made public.

Hearing on Construction Permit -- The next step in the regulatory process is the scheduling of a public hearing to consider issuance of a construction permit for the reactor. The public is given 30 days' notice of the hearing date, through a notice published in the Federal Register. In addition, a public announcement is issued by AEC.

Approximately 20 days in advance of the public hearing, an analysis of the safety aspects of the proposed reactor, prepared by the regulatory staff, is made available to the public. This analysis takes into account the recommendations and advice of the ACRS. Copies of this analysis are furnished to newspapers in the area which surrounds the proposed site of the reactor.

The public hearing is conducted either by an atomic safety and licensing board, or by a hearing examiner. An atomic safety and licensing board is composed of two persons who are technically qualified and one person who is qualified to conduct administrative proceedings.

The application, any amendments to the application which may have been filed, and any other pertinent documents are submitted for the record. Testimony is presented both by the applicant and the AEC regulatory staff on the safety aspects of the reactor and on the applicant's technical and financial qualifications to construct the reactor. In addition, the Commission's rules of practice permit persons whose interests may be affected by the proceeding to intervene as parties. Persons who wish only to make a statement of their views concerning the project may be permitted to make "limited appearances."

The proceedings are conducted informally, consistent with legal requirements and fairness to all parties.

Final Action on Construction Permit -- The licensing board or hearing examiner considers the evidence which has been presented, together with any briefs which may have been filed, and issues a decision. These decisions are subject to review by the full Commission upon its own motion or upon filing of a petition for review which the Commission accepts. In the absence of Commission review, the decision becomes effective on a specified date.

If authorized by the decision, a construction permit is issued. The permit for a power reactor may be provisional in that it takes into account the fact that complete technical data being developed by the applicant will be required later to complete the safety analysis. The permit must include a finding, however, that AEC is satisfied it has enough information to provide reasonable assurance that a facility of the type and size proposed can be constructed and operated safely at the proposed location.

The Commission, as a matter of practice, informally reviews proposed construction permits and operating licenses in cases where no hearing is held.

Operating License -- As construction proceeds on the reactor, it is inspected periodically by representatives of the Commission's Division of Compliance to assure that the requirements of the construction permit are met.

Eventually, the applicant submits the remainder of the required information, including plans for operation and procedures for coping with emergency situations, and pertinent details on the final design of the reactor itself - such as containment design, design of the nuclear core, and waste disposal systems. Once again the regulatory staff makes a detailed review of the information on the reactor and presents a summary of its views to the Advisory Committee on Reactor Safeguards. The report subsequently developed containing advice by the ACRS to the Commission is made public.

When the Advisory Committee on Reactor Safeguards and the AEC regulatory staff have completed their preoperational safety reviews, the Commission in its discretion may issue an operating license (after the required safety determination has been made) or it may schedule a public hearing to consider issuance of the license. In the event no hearing is scheduled, a 30-day public notice is published stating that in the absence of a timely petition to intervene, and request for a hearing, the license will be issued. Before final action on the license, the regulatory staff's analysis of the safety aspects is made available to the public.

If a public hearing is held, the decision of the licensing board or hearing examiner is subject to Commission review.

If the decision is favorable, an operating license is issued to the applicant. This license may be provisional for an initial period of operation, at the end of which time a review is made to determine conditions for the full term license (not more than 40 years).

Power reactors are brought up to full power gradually. In some cases, a step-wise approach to full power operation is required by the license, and an evaluation of operations is made by AEC before each new step in power increase is begun. Persons who operate a power reactor must be licensed by the Commission. They must pass an AEC examination which includes an operating test and a written examination on their knowledge of specific details of the facility and the procedures used in its operation.

Continuing Review -- AEC inspection and review of power reactors does not stop when the operating license is issued. The reactors are inspected periodically by representatives of the Commission's Division of Compliance to make certain operations are being conducted in accordance with terms of the license.

Thus, reactors are subject to detailed review by technical experts before construction is permitted, before operation is permitted and during the entire period of their operation. In the event an unsafe condition is discovered after operation begins, the Commission has authority to order the licensee to shut down the reactor and take any safety measures which may be necessary. It should be emphasized, however, that the outstanding safety record of the atomic energy industry has been achieved because persons who deal with atomic energy respect the potential hazards and exercise great care in the handling and use of atomic materials.

THE REGULATORY STAFF'S TECHNICAL REVIEW

The goal of the regulatory staff's technical analysis and safety evaluation of nuclear reactor proposals is to assure that adequate provisions have been made in their siting, design, construction and operation so that neither normal operation nor accidents will pose any significant risk to the health and safety of the employees or the public.

Of overriding importance in this review is the determination of the adequacy of two general types of safeguards to (1) prevent the occurrence of any credible accident, and (2) in the event that such an accident should somehow occur, to prevent overexposure of the public and employees to radiation.

The site as well as the principal features of the plant and its operation come under scrutiny by the staff in arriving at such assurance. Factors considered in judging the safety of proposed sites include dimensions and characteristics of the site under the operator's control; population density in the area surrounding the proposed site, and the uses which are made of this area, such as industrial, farming or residential; and the seismology, meteorology, geology and hydrology of the area. Utilization is made of the advice of recognized authorities

in the pertinent fields of the project's environment, including Federal agencies such as the U. S. Geological Survey and the U. S. Weather Bureau.

The old adage that "an ounce of prevention is worth a pound of cure" is particularly applicable to the overall safety problem of nuclear power plant operation. First attention is therefore given by the designer, the applicant and the regulatory staff to those facets of reactor design which are essential to stable operation within limits known from experimental and experience information to be adequate to protect the integrity of the nuclear fuel and primary coolant system. The nuclear limits are established for a particular core by a complicated nuclear design procedure, and guaranteed during operation by nuclear control instrumentation with accompanying "scram" devices set well below the points at which fuel damage will occur. Limitations on the pressure, temperature and flow of the primary system coolant are established from engineering considerations, and are guaranteed by comparable "stop" devices set with ample margin to protect the integrity of both the fuel and the primary coolant system. Administrative limitations on operating procedures must be adequate to preclude the likelihood of operational errors that might conceivably defeat the automatic protective devices.

While accident prevention is the cornerstone both of safety and of the economics of nuclear power generation, it is nevertheless necessary to make provisions for limiting the consequences of any accidents that may occur, despite the precautions taken to prevent them. Such accidents will be due primarily to component malfunction or materials failure of a type that cannot usually be predicted. Accident consequence-limiting devices are an important part of every nuclear facility design, and the regulatory staff takes a careful engineering look at the adequacy of this feature in design proposals.

First and foremost is the emergency cooling system which is designed to prevent core meltdown in the event of a major rupture in the primary cooling system that would interrupt the normal coolant flow. The probable reliability of emergency cooling systems over the life of the plant, and methods of periodic checking of their operability come in for close scrutiny by the staff.

If the emergency cooling system works as planned, no significant release of fission products from the core should occur, even with a major rupture in the primary coolant system. But this cannot be taken for granted, and provisions must therefore be made to contain any fission products that may be released from the core in the event of an accident involving both a major rupture of the primary system and the failure or partial malfunction of the emergency cooling devices. There are several different current designs of containment structures, all of which must be evaluated by the staff with respect to their adequacy to accommodate with suitable margins of safety the pressure and energy releases resulting from a major break in the primary system, and to hold the leakage of fission products outside the containment to acceptable limits as determined by resident population patterns around the facility. The criteria used in the evaluation are those set forth in Part 100 of the Commission's regulations.

Where needed to prevent over-pressurization of the containment or to lower the iodine content of the containment atmosphere to meet off-site exposure requirements, two additional engineered safeguards have been proposed by license applicants and must be evaluated by the staff. One of these is a series of air recirculation fans equipped with re-circulating coolant which removes the heat to the outside of the containment structures. Where iodine is a problem, these fans are equipped with particulate filters. Another engineered safeguard is a containment spray system with built-in heat exchanger which serves both to cool the containment air and to wash out some of the iodine. In the pressure suppression type of containment, primary system steam and also iodine are trapped in a pool of water.

In its evaluation of the containment and other engineered safeguards, the regulatory staff concerns itself with such things as pressure capability, leak tightness and means of verification, functioning of air recirculation, filtration and spray systems under accident conditions, and the reliability of emergency power to operate all essential equipment in the event of an accident which might incapacitate normal sources of power.

The disposal to the environment of radioactive wastes generated in the facility operation is limited by provisions set forth in Part 20 of

the Commission's regulations. Nevertheless, the regulatory staff evaluates the proposed equipment and methods of meeting these requirements. Since waste processing storage and monitoring are usually conducted outside the reactor containment, the staff also makes an evaluation of the consequences of accidental releases of waste effluents.

COMPLIANCE INSPECTION

When a construction permit is issued, the inspection program of the Division of Compliance begins, and this surveillance continues throughout the lifetime of the plant.

During the construction phase, the Compliance program is directed primarily toward the gathering and reporting of information to substantiate an eventual finding that the facility has been constructed in accordance with the construction permit, and that the preoperational tests demonstrate that the facility and its component system will perform as described in the license application. In the course of gathering this information, inspectors (1) examine the details of construction;(2) ascertain whether the applicant and his contractors maintain an adequate program of inspection of the fabrication of crucial components; (3) examine the plans for, and results of preoperational tests of components and systems, and witness important tests; (4) review the status of the applicant's proposed detailed operating and administrative procedures; and (5) review the applicant's training program and the overall competence of the operating and technical staff. The review of personnel training and competence is later supplemented by the examination of candidates for operator licenses by the Division of Reactor Licensing.

After the initial operating license is issued, the efforts of the Division of Compliance are directed toward verifying that the facility is being operated in accordance with the terms of the license and AEC rules and regulations, and that no undue hazards exist. On-site inspections are conducted at frequencies dependent upon the relative hazard potential of the particular facility and the type of operations currently being carried on. Inspections are quite frequent for complex facilities during initial operational testing, and much less frequent during routine operation of

standard reactors by experienced personnel. During all inspections, the primary sources of information are personal observation by the inspectors, discussion with operating personnel, and review of operating logs and records. Important aspects of the inspection program include:

(1) Observation of all phases of operation of the facility,

(2) review of operation and administrative records,

(3) study of maintenance plans and performances,

(4) investigation of unusual occurrences at the facility,

(5) evaluation of the adequacy of operating and emergency procedures, and

(6) general review of organization and assignments of responsibility, and an overall assessment of management's views on safety.

In the course of the inspection program, close contact is maintained with responsible levels of licensee management to assure that they are aware of any deficiencies discovered, and to ascertain that adequate corrective measures are undertaken. Appropriate informal and formal enforcement programs are available for use as needed.

THE OPERATIONAL SAFETY RECORD OF NUCLEAR REACTORS

In considering the extension of liability protection afforded by the Price-Anderson legislation, it is appropriate to review the safety experience accumulated in the operation of nuclear reactors and the present outlook.

In summary, the number of reactors built and operated has increased sharply since 1957, the greatly expanded operating experience has been accumulated with an outstanding record of safety, and belief in the extremely low likelihood of publicly hazardous accidents can be held with even greater confidence today than could be supported in 1957.

Reactors Built and Operated

As of June, 1965, a total of 281 reactors has been constructed and operated in the United States.* Representing a wide variety of types and

*Critical facilities, of which 83 cells have been constructed and operated, are not included in this total, though in some of them the loading goes above "critical," and neutron chain reaction does become self-sustaining. In some cases a facility is used both as a critical and as a reactor facility; its principal use determines the category. For example, the Godiva facility is listed as a critical facility, though at times it can be considered a reactor.

sizes, these reactors were built for many different functions, are located at diverse types of sites, and have been operated by widely dissimilar organizations.

In size, these reactors ranged from zero power level to some of several thousand megawatts thermal. Types included pressurized and boiling water, organic, gas-cooled, sodium-cooled, pressure-tube, heavy water moderated, super-heat and others. Their functions or purposes have included production (of fissile material), propulsion (of mobile vehicles, namely, maritime vessels), power (i. e., electric), experimental, research, testing and training.

Reactors have been located as far apart as the two poles, almost literally, and at many different places in between; in submarines, in surface ships, at remote military bases, at the isolated National Reactor Testing Station in Idaho, on college campuses, in an airplane, in cities, at remote sites in the country, in a satellite.

The 98 power, production and propulsion reactors have been operated for a cumulative total of almost 500 reactor years. The remaining 183 experimental, test, research and training reactors have attained a cumulative total of almost 1000 operating years.

Thus, a cumulative total of almost 1500 reactor years of operation has been logged by 281 reactors, collectively. The earliest power reactor, Shippingport, which began operation in 1957, has contributed 8 reactor years. Nautilus, the oldest nuclear-propelled vessel, has accumulated 11 reactor years of operation. The Oak Ridge Graphite Reactor, at the time of its shutdown in 1963, had a longer span in operational status than any other reactor, 20 years.

It is significant to the discussion below that this record includes the experience for all types of reactors operated under widely diverse conditions, for different purposes, and during those "early days" when each reactor operation was a venture into an unknown area, as well the experience in more recent days when reactor technology and practice have become more firmly established.

In considering this 1500 reactor years of operating experience with these machines, which possess a well recognized potential for hazard to people, and with a large percentage of the effort devoted to pioneering exploration into a new technology, it is of crucial interest to inquire about abnormalities in operation, accidents occurring and the general safety record experienced. An outstandingly favorable response can be given to this inquiry, as discussed below. However, it would be contrary to all human experience to expect that all mishaps could be eliminated from the operation of nuclear reactors. As in any complex operation, it is not possible to do this. But, as the record of abnormalities in reactor operation is discussed, it should be kept in mind that the basic concept of protection of the public from the potential hazards arising from nuclear reactors is comprised of a twofold tandem approach:

1. There is a major, concerted and continuing effort to reduce to the lowest possible levels the likelihood of abnormalities and mishaps of all magnitudes.

2. There is a deliberate inclusion of protective safeguard systems with the intention that no hazard would occur to the public even if all sorts of imaginable mishaps should occur.

To the present, the latter objective has been fully accomplished. No reactor accident in the United States has caused injury or even interference with the normal activities of any member of the public.

Reactor Incident Experience

In any complex machinery it is inevitable that structural failures, instrument malfunctions, operator errors and other mishaps will occur, despite the most careful design and rigid schedules of maintenance and administrative control. Such has been the experience with reactor facilities, as with all others. At every reactor facility, a record of operating abnormalities accumulates over a period of time. A complete tabulation for all reactors, should one be compiled, would undoubtedly include thousands of such incidents. Most such abnormalities result in no harmful effects, no injury to people, no damage to the facility. However, in a typical facility some mishaps would lead to reactor shutdown and in a very small percentage of the events some minor damages might be involved. A few events have occurred which would merit an "accident" label.

The fact that many abnormalities have been experienced gives rise to considerable reassurance that the margins of safety surely must be large for this number of mishaps to have resulted in as few events, as noted below, and in no events of magnitude sufficient to have caused damage to the public.

Among the abnormal events in reactor systems are a few incipient failures which, had complete failure or conceivable extensions of the situations occurred, could have resulted in serious accidents - some perhaps more serious than any yet experienced. Even then, however, it is extremely unlikely that all safeguard barriers would have been breached or that any harm to the public would have resulted. As examples, in three reactors, two or three of the stud bolts on the head closure of the main pressure vessel or at crucial locations within the structural core supports were badly cracked or broken. In another reactor two main control rod shafts were found to be cracked from stress corrosion. In two reactors significant cracks were found in the piping of the main primary coolant system.

Some of the more serious events which have occurred in reactors resulted in substantial damage to the facility, exposure of personnel above Part 20 limits or unintended major elevation in power (without damage to the facility or overexposure of people). These incidents, all of which have been previously described in published material when consequences of significance were involved, have in no instance caused hazard beyond the site boundaries. A review of reactor operating records reveals that some 25 such events have occurred in the 281 reactors built and operated in the United States.*

The only fatalities from abnormal events in reactor operation occurred in 1961, when the 3 operators of a small experimental reactor at NRTS were killed; 9 operators in 3 separate reactor incidents received exposures in the range from 2 to 20 r; and in 13 events substantial damage was done to

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*Incidents of this type in critical facilities and in fuel processing systems are not included in this tabulation. Also events in radioisotope handling around a reactor, in fuel handling outside of reactors, and non-nuclear incidents in non-nuclear auxiliary systems are not included.

the reactor fuel. Among the principal causes** of these 25 events are the following: blocked coolant flow channel, improper manipulation of control rods, instrument failures or inadequacy, interaction of experiments with reactor behavior, exploratory operation in known hazardous modes of reactor behavior, and erroneous calculation of rod startup patterns.

The large majority of these events have occurred in experimental, test and flexible reactor facilities. Very few such events have occurred in large reactors designed for extended operation for power or other purposes.

Consequences of accidents thus far have been much less severe than might have been expected from the accident conditions and circumstances.

Nothing approaching an abnormality of "maximum credible accident" magnitude has thus far occurred, except possibly for the experimental SL-1 reactor, which damage occurred in the range of what may have been estimated as the maximum credible accident for this small reactor.

Thus, the overall accident experience record must be considered extremely favorable. Only 25 accidents have occurred in 281 assorted reactors of all kinds, operated for a cumulative total of almost 1500 reactor years. No member of the general public has received overexposure to radiation from reactor accidents. No accident has caused interference with normal activities of the general public.

Nevertheless, these damaging incidents have occurred, in some cases more extensive consequences could be visualized under somewhat different conditions, and there can be no firm assurance that more serious accidents will not occur. Further, it must be recognized that 1500 total reactor years of operation, including 500 reactor years for central station power reactors, do not constitute a very extensive basis for predicting long-term expectancy of major (publicly hazardous) reactor accidents. No such accident has occurred; but no central station power reactor has operated longer than 8 years. The patterns of maintenance schedules and component failures for the latter portion of reactor life cycles have not yet evolved, and the full scope of reactor technology has not yet been fully unfolded.

**In some cases more than one factor "caused" the accident. In these cases, what appears to be the principal cause is listed.

On the whole, however, the cumulative record of reactor operations and safety experience gives increasing confidence that the probability of catastrophic accidents and the likelihood of substantial damages to public health and safety is extremely low, particularly for large steadily-operated power reactors.

Several additional factors also produce greatly increased confidence in the safety of nuclear power plants, and support the belief that the probability of publicly hazardous accidents is indeed small.

PROGRESS IN REACTOR SAFETY RESEARCH

The extensive reactor research programs, which currently are being supported to the extent of many millions of dollars annually for efforts directly committed to exploration of reactor safety, have greatly expanded understanding of basic reactor characteristics and behavior. In many areas of reactor technology, our basic knowledge has been substantially increased during the last 8 years since operation of central station nuclear power plants began:

1. Through power excursion experiments (SPERT, BORAX, etc.), knowledge of the dynamic behavior of reactors in reactivity transients has been increased.

2. The inherent coefficients and parameters which lead to automatic termination of reactivity excursions in water reactors with low enrichment oxide cores are much better understood.

3. Through observation of planned experiments and analysis of a few unplanned incidents, knowledge has been increased on factors affecting potential reactions between water and various metals commonly used in core construction.

4. Through extensive small scale experiments, currently being expanded into experiments utilizing large scale equipment, knowledge has been increased on factors affecting potential fractional release of fission products from damaged reactor fuel.

5. Our understanding of radiation embrittlement of structural components of reactors has been extended somewhat and, more importantly, a massive, long-range program is beginning to yield a necessarily slow, but

valuable accumulation of data on this phenomenon. Factors affecting radiation embrittlement and its reversibility are being systematically explored.

6. Energy releases in potential reactor accidents, the subsequent pressure and temperature transients, and design of structuves to accommodate these have been explored. For example, engineering experiments on scale models have demonstrated that with properly designed containment buildings, "upper limit" energy releases even in fast reactors will not result in breach of the containment by ejection of top shield plug, or other potential missiles. In thermal reactors this is not considered to be a problem.

Much new knowledge about reactor behavior and reliability has also been gained in the successful design and operation of successive "generations" of reactors, particularly the water type, and by experimental variation of components and operating modes in such reactors. Much credit must be given to manufacturers and owners for the contribution to basic understanding of reactors, and hence to reactor safety, which has emerged from this work:

1. The stability, controllability and predictability of behavior of certain types of reactors have been amply demonstrated.

2. The technology of controlling and predicting power distribution in reactors has been greatly advanced.

3. Knowledge of burn-out ratios, safe limits on heat fluxes and hence of factors affecting safety margins in thermal design has been clarified.

4. Technology for achieving efficiency in fuel loading and fuel cycles, made possible by predictability in reactor characteristics and behavior through flux flattening, power density control, burnable poisons, chemical shims, etc., has been greatly advanced with accompanying implications for safety of reactors.

5. The technology of fuel element fabrication, quality control, etc., leading to extended life expectancy, has also given greater confidence in safety.

These increases in the understanding of reactor technology have provided a more substantial basis for designing reactors with confidence in the predictability of their safety and performance. Given further extension of our unfolding knowledge about reactor behavior and the reliability of its components, the remaining problems arising from inadequate experience and uncompleted explorations will be resolved. For example, ways will be found to cope with our presently incomplete understanding of the radiation embrittlement problem, and with our inability to make periodic inspections of reactor internals for incipient failure.

THE DEVELOPMENT OF ENGINEERED SAFEGUARDS

When the Commission's guide on the selection of sites for power reactors was published in 1961, the principle was established that, if adequate and dependable engineering safeguards were included in the design of the reactor, there could be some reduction in distance requirements that otherwise would be necessary. A strong incentive was thereby placed before the reactor manufacturers and owners to develop improved safeguard systems in order to gain approval of less isolated reactor sites. Consequently, many new and fruitful ideas on reactor safeguard systems have been generated, and great strides have been made in improving the effectiveness and reliability of such systems.

In safeguards directed toward the prevention of reactor accidents, much general progress and some specific advancements have been made. For convenience, an "accident prevention safeguard" can be defined as any device or system intended to prevent some mishap -- a breakdown of machinery, component failure or malfunction, operator error, etc. -- from escalation into, say, a significant release of fission products from the reactor fuel. Examples of accident prevention safeguards are:

1. A system for rapid automatic rod insertion or reactor "scram" is provided to prevent an inadvertent reactivity insertion from causing a major power excursion and fuel meltdown.

2. An emergency cooling system is provided to prevent a failure of the primary coolant system from causing fuel overheating and meltdown.

3. A poison injection system, to assist in emergency shutdown.

4. Independent source(s) of main station power, in case the primary source of power should fail.

5. A high pressure core spray system to cool the fuel, to be used in case coolant recirculation lines are inoperative.

Substantial progress in the prevention of accidents has also been made through development of more definitive standards, codes and specifications for the design and construction of reactor pressure vessels, the valves and piping of the primary system, and other vital reactor components.

Perhaps even more obvious advances have been made in safeguard systems directed toward minimizing the hazardous consequences of a major accident, should one occur. No substantial hazards to the public could arise from accidents in a reactor unless significant portions of its radioactivity inventory should escape into the environment. Therefore, in conformity with the definition of accidents given above, "consequences-limiting" safeguards would have as their central purpose the prevention of fission product release to the environment which might have accidentally escaped from the fuel.

Among such safeguards are the following:

1. Improved sources of auxiliary power to keep essential equipment in operation in case of failure of normal and emergency power supplies.

2. Spray-washdown systems to quench steam, reduce air temperature, and wash down fission products in the containment in case of accidents.

3. Filter systems to collect fission products by internal recirculation of building atmosphere.

4. Containment buildings of high strength and low-leakage. A number of improved new containment concepts have been evolved, such as:

   (a) Double-containment-with-pumpback, and extra shielding outside.

   (b) Dry well containment with vapor suppression.

   (c) Double vapor suppression.

Such safeguard systems are now being incorporated into the reactor facilities under construction today. They provide a basis for increased confidence in the level of protection against potential accidents in the operation of nuclear reactors. When such systems have been proven out by accumulated experience with responses to periodic tests, performance under simulated accident conditions, and revisions made in design as may be necessary to satisfy capability and testability requirements, an even firmer basis will have been laid for confidence in the ability of such systems to prevent hazards to the public from any foreseeable accidents. Very great strides have already been made toward this end.

In summary, a number of factors contribute to the basis for our confidence and belief that the probability for catastrophic accidents in nuclear reactors in the United States is extremely low -- even lower than our estimate in 1957 -- of the "remote" probability of such an event.

June 21, 1965

PROGRAM FOR DEVELOPMENT OF FURTHER DATA
RELATING TO REACTOR DESIGN AND LOCATION

In April of 1962 the AEC issued Reactor Site Criteria (10 CFR 100) to guide evaluation of proposed sites for stationary power reactors, primarily of the pressurized and boiling water types. These guides were developed in consultation with the AEC statutory Advisory Committee on Reactor Safeguards and the nuclear power industry. One of the objectives of these guides was to familiarize industry, state and local officials and the general public with the factors considered by AEC in evaluating proposed reactor sites.

In acknowledging the continued need for competent technical judgment by both the applicants and the AEC in implementation, flexibility was deliberately written into the guides. The Commission endeavored to set forth the criteria in such a way that they could be administered in an evolutionary manner. In so doing, it was recognized that insufficient experience had been accumulated to permit the writing of detailed standards for either the design or the siting of reactors.

The basic principle underlying the reactor safety program is recognition of the necessity for effective engineered safeguards in the construction and operation of reactors both to prevent major accidents and to control their consequences in the unlikely event they should occur. Under the Site Criteria,* provision is made to balance such engineered safeguards in relation to the distance between reactor and population centers. The application of this concept has had the effect of continuing the AEC practice of keeping central station power and test reactors

* See attachment #1 for digest of Site Criteria.

a reasonable distance from densely populated centers.

The safety record achieved by the nuclear power industry has been outstanding. In no instance has a reactor accident in the operation of a central station electrical generating reactor plant caused any radiation damage to the public or its environment, or to the plant workers. These operations in the United States have involved the production of over 14 billion KW hours of electrical energy.

This exemplary safety record is the result of the general recognition by the Commission and the industry of the problems of potential accidents; the expanding knowledge of phenomena associated with such potential accidents; and the application of sound engineering principles in the design and construction of reactor systems and appurtenances. The Commission's research, development and test programs have materially assisted designers in specifying proper system requirements and safety evaluators in assessing the safety of specific system designs. One part of the Commission's research and development program identified as Nuclear Safety is budgeted at about $24 million in FY 1966.*

The participation of industry has been most encouraging and many recent and improved safety systems have been developed through their efforts; more definitive and uniform criteria for the construction and operation of power reactors have evolved. Efforts along these lines are continuing and are recognized as vital to the healthy growth of the industry. Consequently, it has been possible in recent years to accept certain engineering safeguards as a partial substitute for the amount of geographic isolation formerly required.

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* See attachment #2 for a summary of this Nuclear Safety Program.

The rapid expansion and development of the nuclear power industry involving different reactor types and increased reactor size for broader applications (e.g. desalting), and the incentives to locate in closer proximity to metropolitan load centers, have focused attention on the continuing need for careful attention to all matters which potentially could affect the health and safety of the public. Consequently, further important advances in reactor plant design, in the capability of safety systems and engineered safeguards, in adapting critical components and systems to accommodate their inspection and testability, and in practical demonstration of dependability of performance of such critical systems, must evolve to keep pace with the development of the nuclear power industry.

In recognition of these increasing needs, the Commission has decided to augment efforts and redirect emphasis to define and develop the improvements in reactor plant design and capability of critical systems and engineered safeguards. This effort will be carried on in cooperation with industry and in conjunction with the Commission's safety research and development programs in order to obtain the accumulation of meaningful experience with respect to capability and reliability of important safety systems. In the course of this undertaking, new and improved methods must be developed for design, construction and operation of central station reactors from the standpoint of safety, including means of testing and inspecting the safety aspects of important systems. Additional efforts also will be undertaken, in cooperation with appropriate groups, in the development of standards and codes for the design and construction of reactors and their principal components.

This research and development work, together with increased emphasis on the development of more specific reactor standards will be necessary as reactors increase in size and are built closer to metropolitan load centers.

As the augmented reactor safety program progresses, the Commission will continue to consider applications for reactor construction permits and operating licenses on a case-by-case basis. As new data and improved safety devices are developed, they will be taken into account in the regulatory process. In recognition of this evolutionary development, the AEC has encouraged applicants and industrial organizations to come in and obtain informal review of their siting problems and reactor plant and component design. In this manner, maximum advantage can be taken of the current knowledge and proposed plans as these may affect the health and safety of the public and the growth of the nuclear power industry.

ATTACHMENT NO. 1

DIGEST OF SITE CRITERIA

The Atomic Energy Commission published "Reactor Site Criteria" Part 100, Title 10, Code of Federal Regulations, as an effective regulation on May 12, 1962.[1/] The fundamental purpose of Part 100 is to provide guides for the evaluation of the suitability of proposed sites for stationary power and testing reactors. The regulation specifically recognizes that insufficient experience has been accumulated to permit the writing of detailed standards and emphasizes that Part 100 is intended as an interim guide to identify a number of factors considered by the Commission in evaluating reactor sites.

The particular factors considered in determining the acceptability of a site are the characteristics of reactor design, including power level, safety features and similar design considerations, population density and use characteristics of the environs, physical characteristics of the site, and the engineered safeguards designed to minimize the consequences of any accident.

The point of departure for the evaluation of site suitability and plant design is the concept of the "maximum credible accident." This represents the upper limit of practical hazard to the public which can be expected from the proposed facility.

The guides make explicit, and define, the concepts of population zones around a reactor which had been observed in practice all along by the Commission; the exclusion zone, where people, if any, are highly mobile and under the direction of the reactor operator in emergencies; the next succeeding low population zone, where evasive or protective measures for people could be taken in case of dangerous releases from the plant; and the city distance, where people are not very mobile, but the distance is sufficient to prevent the worst aspects of danger should a serious accident occur.

1/ 27 F.R. 3509

The guides provide numerical values for potential radiation exposure doses that might occur to off-site personnel in case of a maximum credible accident for the reactor in question, to be used as measuring indices for evaluating the acceptability of the reactor in the site proposed.* These values are 25 rem whole body and 300 rem to the thyroid.**

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* These values are not permissible limits for public exposure and are not guides for determining measures that should be taken in case an accident should occur to limit exposures of off-site personnel. They are measuring indices to be used in the context of maximum credible accidents to evaluate reactor designs and reactor sites. To the extent that the protective action guides recently developed by the FRC are applicable, it is assumed that these would be used by persons responsible for deciding protective measures to be taken after an accident.

** In view of the extremely small probability that an accident would occur of the type that would result in such exposures, and in view of the existence of emergency procedures which would reduce still further the likelihood of such exposure, the use of these values in evaluating reactor sites are considered to represent extremely small risks to individuals. Further, calculations show that for presently used reactor sites, these exposure values, in the unlikely event that serious accidents should occur, would result in cumulative total population doses well below the limits which have been recommended by the International Committee on Radiological Protection and others for such population exposure doses.

The guides identify problems of meteorology, hydrology and seismology which must be considered in evaluating the suitability of a site.

Finally, the principle is articulated that "where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards." This has been interpreted to mean that where appropriate and adequate compensating safeguards are provided, there may be some reduction in the isolation distance of the reactor which otherwise would be required.

ATTACHMENT NO. 2

SUMMARY OF NUCLEAR SAFETY PROGRAM

In augmentation of selected results from individual reactor projects and technical programs, the nuclear safety research, development and test program coordinates the technology, focuses effort and provides data and information required for the location, design, construction and operation of safe reactors and the application of reliable engineered safeguards for effective accident prevention and control.

The primary objective of the nuclear safety program is to provide sound technical bases for designing safe reactors and preventing accidents by obtaining quantitative data from a coordinated sequence of laboratory studies and experiments and engineering scale tests, and through improved analytical techniques and processes. Concomitantly, the program's objective is also to provide the required technical bases for evaluating the safety of proposed reactor designs and installations.

The Nuclear Safety Program includes the following:

1. Nuclear safety research and development, which involve both fundamental and applied research in nuclear reactor safety, including studies in reactor kinetics, fuel meltdown phenomena and fuel-coolant interactions (chemical reactions), reactor containment, and other engineered safeguards, including those related to fast

reactor safety studies.

2. Effluent control research and development, which is directed toward the development of more effective means of managing radioactive wastes resulting from nuclear reactor operations, and toward determining and controlling the behavior of these residual radioactive effluents in the environment.

3. Engineering field tests, which extend laboratory-scale test results on both aerospace and terrestrial nuclear systems into full-engineering-scale field test results.

4. Reactor safety analysis and evaluation, which conducts safety analysis studies that supplement other program activities, and provide assistance in planning and directing the over-all nuclear safety program.

The kinetics program is characterized by the SPERT reactors which involve study of all phases of abnormal reactor behavior including nuclear excursions beyond the threshold of damage.

Reactor containment studies include large scale experiments of energy releases, the efficiency of engineered safeguards, fission product release and control, and pipe rupture.

Work carried out under effluent control research and development includes environmental studies associated with the safe disposal of radioactive effluents into the environment including gaseous activity, low, intermediate and high activity waste disposal, the

development of waste treatment processes, and the development of countermeasures for trapping fission products resulting from a reactor accident.

The engineering test program is characterized by the LOFT program which involves the test and control of a full scale loss of coolant accident for a 50 Mw(t) pressurized water reactor. In many respects, this program is the synthesis of and focus for other parts of the safety program. The aerospace safety program is also conducted under this activity.

The nuclear safety analysis and evaluation effort involves the coordination of safety related work outside of the base safety programs, pressure vessel research, safety analysis of reactors, surveys of reactor operating experience and adequacy of engineered safeguards, probabilistic studies of reactor accident occurence, and nuclear safety data dissemination.

Currently, an important part of the analysis and evaluation work is the initiation of development of more uniform and better engineering codes, material and component specifications and quality control and inspection practices. It is intended that ultimately there will be a comprehensive set of codes or specifications for all components of reactor systems that, if properly implemented and followed through, will maximize the safety of reactors at any

location, both in terms of protecting the public and maintaining the integrity of the operating plant.

The budget for this effort is given in the following estimates for FY 1966 and FY 1967.

Budget Estimates (In Millions)

	FY 1966	FY 1967
Research and Development	10.0	15.0
Effluent Control R&D	7.0	7.0
Engineering Field Tests	18.0	22.0
Analysis & Evaluation	1.0	2.0
	36.0	46.0