COMMONWEALTH EDISON COMPANY

GENERAL ELECTRIC COMPANY

25,745

GEAP-0878

MASTER

ENCLOSURE SECTION

OF THE

HAZARDS SUMMARY REPORT

FOR THE

DRESDEN NUCLEAR POWER STATION

By: Commonwealth Edison Company

General Electric Company

Submitted in accordance with provisions of

Construction Permit No. CPPR-2

June 12, 1957

003 001

RAndolph 6-1200

COMMONWEALTH EDISON COMPANY

72 WEST ADAMS STREET • CHICAGO 90, ILLINOIS

June 12, 1957

United States Atomic Energy Commission
Division of Civilian Application
Washington 25, D. C.

Attention: Mr. Harold L. Price, Director

SUBJECT: ENCLOSURE SECTION OF THE HAZARDS SUMMARY REPORT

Gentlemen:

Commonwealth Edison Company for the purpose of supplementing the license application for the Dresden Nuclear Power Station contained in its proposal addressed to you under date of March 31, 1955, and to comply with paragraph 4, page #2 of the Construction Permit #CPPR-2, submits herewith a portion of the Final Hazards Summary Report entitled Enclosure Section.

Respectfully submitted,

COMMONWEALTH EDISON COMPANY

By R. D. Maxson
R. D. Maxson
Senior Vice President

Submitted and sworn to before me
this 12th day of June 1957
by said R. D. Maxson

Notary Public
Cook County, Illinois

My commission expires:

January 4, 1961

Safeguards Unit

GEAP-0878

(Revised June 3, 1957)

ENCLOSURE SECTION
OF THE
HAZARDS SUMMARY REPORT
FOR THE
DRESDEN NUCLEAR POWER STATION

Prepared by: R. C. Thorburn
C. A. Pursel
G. Sege

Submitted by: G. Sege
G. Sege
Manager - Safeguard Evaluations
Power-Reactor Engineering

Approved: J R Wolcott
J. R. Wolcott
Manager -- Power-Reactor Engineering

V D Nixon
V. D. Nixon
Manager -- Commonwealth Edison Project

January 24, 1957

ATOMIC POWER EQUIPMENT DEPARTMENT
GENERAL ELECTRIC COMPANY

PART ONE

SUMMARY

The General Electric Company is designing and building a 180,000 kilowatt nuclear power plant for the Commonwealth Edison Company at a site near the confluence of the Kankakee and Des Plaines Rivers in Grundy County, Illinois, about 47 miles southwest of Chicago.

The plant will be known as the Dresden Nuclear Power Station, and will employ a nuclear reactor of the dual-cycle boiling water type. The general features of the reactor, the associated power plant, and the site on which the plant will be located have been described in previous submittals to the Atomic Energy Commission.

A construction permit for the plant was issued to the Commonwealth Edison Company on May 4, 1956. This permit, CPPR-2, states, among other things, that the Commission has found that a facility of the general type proposed can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

The permit is subject to submittal to the Commission of a Final Hazards Summary Report (portions of which may be submitted and evaluated from time to time) and a finding by the Commission that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in accordance with the specified procedures.

This is the first of a series of reports which will be submitted to the Commission as portions of the Hazards Summary Report in compliance with these provisions.

The following material has been prepared by the General Electric Company as the Enclosure Section of the Hazards Summary Report for the Dresden Nuclear Power Station.

The purpose of this report is to describe, with particular emphasis on safety aspects, the general design features of the gas-tight, spherical steel enclosure, 190 feet in diameter, which will house the reactor and its associated equipment, and to outline the bases on which these features were selected. The report requests the concurrence of the Commission with the considerations and conclusions set forth in the text.

The spherical enclosure is a structural supplement to the inherent safety of the Dresden boiling water reactor and to the safety devices provided to control the machine in case of emergencies. It is designed so that it is capable of confining any radioactive vapors that may be liberated from the reactor in the very unlikely event of an accident.

The general design features of the spherical steel enclosure as described in the report are:

1. The enclosure will house the reactor, primary recirculation piping, pumps, steam drum, steam generators, and emergency heat exchangers, but not the turbine-generator.

2. The enclosure is designed to withstand the internal pressures which would result from the most severe incident it is reasonable to conceive. Substantial safety factors, based on the A. S. M. E. Boiler and Pressure Vessel Code, were employed in designing the enclosure.

3. The enclosure is designed for leak tightness. Radiographing of all welds and seams, and pneumatic pressure tests will assure that all imperfections are corrected and that the structure meets design specifications.

4. The enclosure is designed to withstand the stresses which may result from earthquake or weather.

5. The enclosure design provides special provisions for assuring the integrity of the enclosure in case of accident.

6. The enclosure will be located at least one-half mile from any population or non-plant structures in the area.

Other items considered in the enclosure design in providing for the safety and protection of the public and which are discussed in the report include:

1. The housing of the turbine-generator and related equipment in a conventional building separate from the sphere, which minimizes the possibility of damage to the enclosure as a result of turbine-generator accident.

2. The studying of the need for any additional protection of the sphere from fragments that could hurl against the enclosure wall as a result of reactor or other equipment accident to the end that the ultimate design of the enclosure will have adequate missile protection.

3. The designing of the plant so that penetrations of the enclosure wall are held to an absolute minimum.

4. The establishing of step-by-step procedures for constructing and testing the reactor enclosure.

The text of the report substantiates these major design features and considerations as stated above, and discusses the assumptions and analyses on which they were based.

PART TWO

TECHNICAL PRESENTATION

I. INTRODUCTION

The General Electric Company is designing and building a 180,000-kw dual-cycle boiling-water-reactor nuclear power plant for the Commonwealth Edison Company. The plant, to be known as the Dresden Nuclear Power Station, will be located at a site near the confluence of the Kankakee and Des Plaines Rivers in Grundy County, Illinois, about 47 miles southwest of Chicago. The general features of the reactor and the associated power plant and the site and its environment have been described in previous submittals to the U.S. Atomic Energy Commission.[1] [2] [3]

On the basis of these submittals a construction permit (No. CPPR-2) for the plant was issued on May 4, 1956. The construction permit states that the Commission has found that a facility of the general type proposed "can be constructed and operated at the proposed location without undue risk to the health and safety of the public."

The permit is subject to submittal to the Commission of a "final Hazards Summary Report (portions of which may be submitted and evaluated from time to time) and a finding by the Commission that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in accordance with the specified procedures".

This is the first of a series of reports contemplated for submittal as portions of the Hazards Summary Report in accordance with the provision indicated in the above paragraph. It is currently planned that subsequent submittals will be as follows:

1. Preliminary Hazards Summary Report.

2. Operating Specifications.

3. Disaster Plan.

4. Final Hazards Summary Report (superseding the preceding submittals and any revisions of them).

Although, in view of the inherent safety features of the reactor and the safety devices provided [1], [2], the possibility of a nuclear accident to the plant is extremely remote, the reactor and associated equipment will be housed in a vapor-tight enclosure to confine the radioactive vapors that might be liberated from the reactor in the event of accident. It is the purpose of this report to describe the general design features of the enclosure and to outline the bases on which these features were chosen. Concurrence of the Atomic Energy Commission with these general features and bases is sought. Specifically, concurrence is sought with the following:

1. Use as the enclosure type of a largely above-ground spherical steel shell housing the reactor, the primary recirculation loop, and an emergency heat exchanger. The turbine will be housed in a separate building of more conventional design.

2. A design pressure -- employing A.S.M.E. Boiler and Pressure Vessel Code, Section VIII safety factors -- that equals or exceeds the internal pressure that would be created by release (and thus partial flashing to steam) of the pressurized hot water which under normal operating

conditions is contained in the reactor and the primary recirculation loop; provided that

a. the ability of the sphere to withstand a pneumatic test of 1.25 times the design pressure is demonstrated, and that

b. this test pressure equals or exceeds that estimated to result from the most severe reactor-rupture accident to which it is reasonable to assign a possibility of occurrence, such a "worst reasonable accident" * being conceived of as involving the maximum reasonable energy contributions from a nuclear excursion and from chemical reaction between reactor components -- in addition to the above-described hot-water liberation.

3. Location of the sphere on the plant site so that it is at least 0.5 mile from

a. Skinner Island in the Kankakee River,

b. the navigation channel in the Illinois River, and

c. the south and west land boundaries of the site.

---

* See Page 26 for explanation of "probably worst" and "worst reasonable" accidents.

4. A degree of leak tightness attained by correction of any imperfections found by radiographing of all welds, leak testing of all welds and seals, and leakage-rate determination for the completed sphere at approximately the design pressure. It is expected that the leakage-rate determination will demonstrate a degree of leak tightness corresponding to not more than approximately 0.5%/day leakage at 1.25 times the design pressure*.

5. Weather, earthquake, and other stress allowances in accordance with the A.S.M.E. Boiler and Pressure Vessel Code, with special considerations as summarized in Section H of this report.

The enclosure constitutes an added measure of nuclear safeguard, supplementing the protection afforded by the inherent safety of the basic reactor design and the safety devices with which it is planned to equip the reactor. The other phases of the system of safeguards against nuclear accidents will be described and evaluated in reports to be submitted later, as indicated above.

* Potential radiological effects of such leakage are indicated in Section E.

II. BASIC FEATURES OF THE ENCLOSURE

The following is a summary of the basic design features of the enclosure for the Dresden Nuclear Power Station reactor. The technical background upon which the selection of each of these features is based is indicated later in this report.

The technical information presented reflects the best thinking and judgment at this writing. In the event that any new information pertinent to the enclosure should be developed in the course of detailed design, such new information will be forwarded.

The enclosure housing the reactor is a spherical steel shell, 190 feet in diameter, largely above ground (with the equator approximately 56 feet above ground level).

Equipment in the primary recirculation loop, an emergency heat exchanger, and certain other equipment are located inside the enclosure along with the reactor. The turbine-generator, the associated condenser, and various other equipment are in a separate building, outside the enclosure.

The design pressure of the enclosure is 29.5 psig, employing the structural safety factors called for by the A.S.M.E. Boiler and Pressure Vessel Code, Section VIII*. Coincidence of a 250°F temperature rise with the occurrence of the design pressure is allowed for. The design pressure equals or exceeds the internal pressure that would be created by the most probable kinds of potential reactor-rupture accidents. These potential accidents are conceived of as consisting in release (and thus partial flashing to steam) of the pressurized hot water contained in the

---

* I.e., maximum allowable membrane stress of 1/4 of the minimum tensile strength, with qualifications as set forth in the Code.

reactor and the primary recirculation loop (primary steam drum, primary side of secondary steam generator, and pumps and piping in the loop) under normal operating conditions, with no significant energy contributions from any nuclear excursion or from any chemical reaction between reactor components. Most potential accidents, those with the highest relative probability of occurring, are considered to be within this class.

The ability of the enclosure to withstand a pneumatic test at 1.25 times the design pressure is to be demonstrated. This test pressure equals or exceeds that estimated to result from the most severe reactor-rupture accident to which it is reasonable to assign a possibility of occurrence. Such a worst reasonable accident is conceived of as involving the following:

a. liberation of the pressurized hot water present in the reactor and primary recirculation loop during normal operation;

b. a nuclear excursion contributing an amount of energy that would be enough to melt all the (uranium oxide) fuel in the reactor core; and

c. chemical reaction of 25% of the zirconium in the fuel-element cladding with water.

The enclosure is designed to withstand weather and earthquakes, as discussed in Section D of this report.

To insure the greatest practical degree of leak-tightness (as well as structural integrity), all welds on the sphere will be radiographed during construction and all welds and seals tested for tightness with soap bubbles. Any imperfections found will be corrected. In addition, adequate tightness of the sphere will be confirmed by determining the leakage rate at a 29.5-psig pneumatic pressure.

The sphere is to be located approximately 0.6 mile south of the navigation channel in the Illinois River and 0.5 mile west of the west bank of Skinner Island in the Kankakee River, the off-site land nearest to the sphere*, as shown in Figure 1.

---

* Other than approximately 0.07 square mile of State-owned flowage land located between 0.3 and 0.5 mile from the sphere.

III. DISCUSSION

A. REACTOR SYSTEM

The dual-cycle boiling reactor is a thermal-neutron heterogeneous reactor employing slightly enriched uranium as the nuclear fuel, moderated and cooled by light water. The cooling water boils in the reactor vessel, which is pressurized to 1,000 psig. The steam thus generated is sent, after moisture removal, directly to the high-pressure admission point of a dual-admission steam turbine. Additional steam is produced in four secondary steam generators heated by hot water withdrawn from the reactor through a steam separating drum and then returned to it. This lower-pressure (approximately 495 psig) steam is routed to the low-pressure (second) admission point of the turbine. A schematic flow diagram is shown as Figure 2. A perspective view of the reactor and primary coolant system is sketched in Figure 3. For a discussion of the general functioning of the reactor and its power-extraction system reference is made to documents NPG-111[1] and X-GEAP-045[2], submitted previously.

The reactor vessel is approximately 42 feet high and 12 feet, 2 inches in internal diameter. It is made of carbon steel clad inside with stainless steel. The shell thickness opposite the nuclear-fuel core is 5-5/8 inches (including the 3/8-inch cladding). The principal features of the reactor design are depicted in Figure 4.

During normal operation the reactor and the primary recirculation loop (steam drum, primary side of the secondary steam generator, pumps, and piping) contain 188 tons of hot water. (Of this, 48 tons are in the reactor itself.)

It is currently planned to use fuel in the form of approximately 17,000 nine-foot-long 0.5-inch-diameter vertical rods of uranium dioxide, encased in Zircaloy-2 cladding. The fuel elements are arranged in seven hundred and twelve 3.8-inch square cross-section flow channels separated by Zircaloy-2 sheeting. The weight of uranium dioxide in the fuel elements is 63 tons; that of Zircaloy-2 in the cladding, 15 tons.

The arrangement of the reactor and other major equipment in the enclosure and turbine building is sketched in Figure 5. The equipment layout within the enclosure is indicated in more detail in Figures 6, 7, and 8.

The principal plant-design parameters are tabulated in Appendix G.

B. ENCLOSURE TYPE AND SIZE

1. General

An above-ground spherical steel shell housing the reactor but not the turbine was selected as the enclosure type on the basis of extensive studies of the safety and other merits of alternative structure types. The studies, conducted by General Electric in collaboration with the Bechtel Corporation, engineer-constructors for the plant, embraced studies of a wide variety of potential designs:

a. Above-ground and underground structures.

b. Steel, concrete, and combinations of the two as structural materials.

c. Spherical, cylindrical, and combination (spherical-and-cylindrical) shapes.

d. Enclosures alternatively housing and not housing the turbine along with the reactor.

A highest practical degree of structural integrity and reliability was sought as the essential requirement and -- with safety comparable -- an economical and convenient housing was desired. While it is recognized that experience with large structures of this kind is limited and that therefore many important factors had to be weighed by considerations necessarily quite speculative, the enclosure type chosen was judged to be best suited to the requirements.

The enclosure size (190-ft. diameter) was determined by the space requirements for the equipment and accessories enclosed and by considerations of reasonable convenience of layout of that equipment.

2. <u>"Reactor-Only" Enclosure</u>

The choice of a "reactor-only" enclosure (i.e., one not housing the turbine-generator along with the reactor) was based on the safety comparison summarized below and on the following other considerations:

a. it results in a more convenient layout;

b. by housing the turbine in a more nearly conventional building, it renders the plant more amenable to future modifications -- an important consideration in a developmental project;

c. for equivalent safety, the reactor-only enclosure is economically the more advantageous; and

d. by reducing layout and structural complexities, it represents a trend desirable from the standpoint of the long-range development of the atomic power industry.

From the nuclear-safety standpoint too the reactor-only type of enclosure offers advantages:

a. The danger of damage to the enclosure in the event of a turbine explosion is minimized with the turbine outside, rotating in a plane not intersecting the enclosure.

b. No penetration through the enclosure wall is required for the generator main leads. This is a particularly vulnerable type of penetration, involving a large-diameter disc of non-magnetic material and temperature differentials due to eddy currents.

c. The danger of damage to the enclosure by explosion of the hydrogen used to cool the generator is eliminated.

d. With less lubricating oil in the enclosure, the fire safety is improved.

e. The total area of penetrations is decreased by approximately 30%.

f. There are fewer man-hours spent inside the enclosure, unprotected in the event of nuclear accident.

g. With less equipment enclosed, the need to cut temporary openings into the shell for removing or bringing in large equipment pieces is likely to be less frequent.

h. The enclosure will have several minor advantages by virtue of its smaller size.

While individually these are trivial, together they are perhaps significant. These small-enclosure advantages are:

(i) Easier to "shadow-shield" after an accident.

(ii) Safer against a tornado.

(iii) Smaller bomb target.

On the other hand, the reactor-only enclosure also has disadvantages from the safety standpoint:

a. Valves in the reactor-to-turbine steam lines must be relied on to complete the enclosure in the event of an accident.

b. Accidental closure of these valves would separate the reactor from its normal heat sink -- the turbine condenser, thus necessitating special provisions to remove the reactor heat by other means.

These safety drawbacks can, however, be reduced to a minimum by proper design, with the net effect that the over-all safety is not less than that attainable with the turbine also enclosed

3. Missile Protection

It is recognized that the value of the enclosure for confining radioactive vapors that might be liberated in an accident would be greatly impaired, perhaps lost, if the enclosure were to be punctured by reactor or other fragments that an accident could hurl against the enclosure wall. Studies are currently under way with the objective of assessing the nature and magnitude of this missile danger, in order that appropriate protection might be incorporated in the design of the plant.* The status of this work is indicated in Appendix E.

4. Insulation

The enclosure will have thermal insulation on the outside, for operating reasons. This will also aid in protection of ductility in cold weather and will provide corrosion protection of the exterior side of the enclosure shell. (The type of insulation has not yet been chosen.)

5. Ventilation

The enclosure will be ventilated on a "once-through" air flow basis, with the cooling effect of the ventilation air supplemented by coolers in the shielded equipment cells.

Ventilation and cooling are being designed to permit personnel access to the enclosure during operation without excessive radiation or temperature exposure and to prevent heat damage to any equipment.

The spent ventilation air will be discharged through a 300-foot-high stack. It will be continuously monitored for radioactivity, to assure that maximum permissible concentrations are not exceeded.

6. Penetrations

The various penetrations through the enclosure wall (pipes, ducts, refueling opening, access locks, electrical leads) will receive special consideration in design from the standpoints of the requirements of leak tightness and prevention of excessive stresses.

Two general types of provisions are contemplated to prevent escape of excessive radioactive material through penetrations in the

---

* Studies are currently being conducted by Broadview Research and Development, Burlingame, California, under contract with General Electric.

event of accident:

a. Penetrations which are closed during normal operation will be protected against being opened during operation, or in potentially hazardous non-operating situations, by appropriate control-instrument interlocks, operating rules, or a combination of the two.

b. Each penetration open during normal operation will be automatically closed in the event of accident fast enough and tightly enough to prevent escape of more than a small fraction of the maximum acceptable total leakage.

Since details of the provisions for insuring closure and tightness of penetrations in the event of accident are not involved in the design and construction of the enclosure itself, detailed description and analysis of these provisions is beyond the scope of this report.

Figures 9 and 10 show possible penetration locations and some of the various alternative designs under current study for the several penetration types.

7. Power Supply

The sources of power supply to the enclosure interior are listed in Appendix F.

8. Fire Protection

Fire protection provisions in the enclosure will be designed in accordance with the standard fire protection requirements of the Commonwealth Edison Company. The provisions will comprise fire hoses and movable (cart-type) carbon-dioxide fire extinguishers.

C. BASIS OF INTERNAL PRESSURE

The enclosure design is based on adequacy for pressures that could be developed in severe terminal accidents, as discussed below. It is emphasized, however, that the possibility of occurrence of any such accident is extremely remote.

The analysis of accident pressures discussed below distinguishes between two limiting-accident-magnitude concepts:

a. A "probably worst accident", creating a pressure which would in all probability not be exceeded in any accident. (Discussed under 1, below.)

b. A "worst reasonable accident" represents the worst accident to which it is reasonable to assign a possibility of occurrence. Such an accident, though even more extremely improbable than the "probably worst" case, cannot be discounted altogether in conservative design. (Discussed under 2, below.)

Hypothetical accidents worse than these, with no significant possibility of occurrence, are also discussed (under 3, below), to illustrate the safety margin available in the design.

1. <u>"Probably Worst" Post-Accident Pressures</u>

The internal pressure in the enclosure in the event of a major accident could be created by contributions from three different potential sources of energy:

a. the pressurized hot water in the reactor and in those of its auxiliaries which are not separated from it by a solid barrier;

b. a nuclear excursion; and

c. chemical reaction between reactor components, viz., between water and the zirconium cladding in which the uranium-oxide fuel is encased.

The hot-water contribution (a) is the most important of these; the nuclear and chemical contributions (b) and (c) would in fact probably be negligible in comparison.

Accordingly, the design pressure of the enclosure, 29.5 psig, has been determined by the pressure that would be created by release (and thus partial flashing to steam) of the pressurized hot water contained in the reactor and the primary recirculation loop (primary steam drum, primary side of secondary steam generators, and pumps and piping in the loop) under normal operating conditions, with no significant nuclear or chemical energy contributions. Under these conditions -- i.e., with the reactor at its rated working pressure (1000 psig), in a boiling condition, operating at its rated power level (630 thermal Mw) -- a reactor rupture would expose 188 tons of pressurized hot water, and the ensuing partial flashing to steam would create a 23-psig pressure in the enclosure. (The 23-psig post-accident pressure was calculated on the basis of an instantaneous release. In the event that the release should take a significant finite time the ultimate pressure reached would be less, because the net heat loss from the vapor space inside the sphere to the sphere shell, to the structure inside the sphere, and to the environment would initially be faster than the heat gained due to radioactive decay.)

The 29.5-psig design pressure will be withstood by the enclosure with the same structural safety factors as are called for by Section VIII of the A.S.M.E. Boiler and Pressure Vessel Code*, for long-continuing and repeated loadings, so that the

---

* I.e., maximum allowable membrane stress of 1/4 of the minimum tensile strength, with qualifications as set forth in the Code.

enclosure will provide a very high degree of reliability for confining the radioactive materials liberated in these most probable accidents.

It is possible that, as the detailed design of the plant is established, there will be some minor changes in the amount of water in the reactor system and in the percentage of unoccupied volume in the sphere. Corresponding to any such changes there would be minor upward or downward changes in the 23-psig calculated accident pressure cited above. However, it is not anticipated that any such changes will raise the calculated "probably worst accident" pressure above the 29.5-psig design pressure.

2. "Worst Reasonable" Post-Accident Pressure

The ability of the enclosure to withstand a pneumatic test of 1.25 times the design pressure (i.e., 37 psig) is to be demonstrated. This test pressure equals or exceeds that estimated to be given rise to by the most severe reactor-rupture accident to which it is reasonable to assign a possibility of occurrence. Such a "worst reasonable accident" is conceived of as involving the following:

a. liberation of the pressurized hot water present in the reactor and primary recirculation loop during normal operation;

b. a nuclear excursion contributing an amount of energy that would be enough to melt all the uranium oxide fuel in the reactor core; and

c. chemical reaction of 25% of the zirconium in the fuel-element cladding with water.

Such a "worst reasonable accident" (assuming the water release to be practically instantaneous) would give rise to an enclosure pressure of 35 psig.

Minor changes in the amounts of water, uranium oxide, zirconium, or sphere free volume in the course of detailed design could change the 35-psig pressure calculated for the "worst reasonable accident" slightly upward or downward. However, it is not anticipated that it will exceed the 37-psig enclosure test pressure.

3. Pressures for Accidents Beyond Reasonably Conceivable Magnitudes

It is possible to conceive hypothetical combinations of circumstances so extreme that it would not be reasonable to ascribe to them a significant possibility of occurring which could give rise to pressures exceeding those created by the "worst reasonable accident" described above. However, the structural safety factors employed in the design of the enclosure give it a good chance of withstanding pressures significantly in excess of its test pressure (which is based on the "worst reasonable accident").

For example, suppose that the chemical reaction involved in the "worst reasonable accident" is augmented by burning of the liberated hydrogen* and proceeds with such violence that missiles formed in the explosive reaction rupture the secondary steam generator, so that the pressurized hot water on its secondary side, as well as that on its primary side, becomes exposed to partial flashing to steam**. The enclosure pressure after

---

* Very unlikely, as discussed below.

** The possibility of such rupture is remote, because the secondary steam generators are 30 feet away from the reactor and there are 12 feet of concrete wall thickness in between.

such an accident would be 39 psig.

As an extreme illustration, suppose that the nuclear and zirconium-water reaction contributions were twice the worst reasonable ones described above and that in addition the secondary steam generator ruptures and the hydrogen liberated in the zirconium-water reaction burns. Such an accident would give rise to a 55-psig pressure. Calculations show that the enclosure should withstand a pressure even as high as that developed in this over-pessimistically conceived case. Although calculations of this kind are subject to uncertainties, this hypothetical illustration has been presented to indicate the extent of safety margin available over the lower pressures that might arise in the earlier-discussed accidents of reasonable magnitudes.

4. Enclosure Pressures After Various Accidents

Enclosure pressures after various potential accidents are indicated in Table I. Figure 11 is a plot of sphere pressure as a function of the amount of pressurized hot water exposed and of the amounts of nuclear and chemical energy contributions. The plot may be used to determine post-accident pressures not specifically considered in Table I.

It should be noted that the tabulated and plotted accident conditions represent highly pessimistic "bench-marks", as discussed above. The more severe of the cases are not assigned any significant possibility of occurrence, but are considered only to delineate the degree of conservativeness of the design.

5. Basis of Maximum Reasonable Nuclear Excursion

The magnitude of credible potential nuclear excursion is difficult to estimate, because any credible combinations of circumstances that might lead to a nuclear excursion will be specifically designed against.

However, disregarding consideration of any specific mechanisms that could lead to such an excursion, it appears reasonable to suppose that the excursion energy would not exceed that which would be just enough to melt all of the uranium-oxide fuel in the core. This comes to 48 million B.t.u. and has been considered as the maximum reasonable nuclear contribution to the post-accident internal pressure.

It is not suggested that melting of the entire core would be the actual mechanism whereby the core is disrupted, thereby terminating the nuclear reaction. Vaporization of the highest-neutron-flux (central) portions of the core while the lower-flux regions are still in a solid state appears to be a more likely mechanism of disruption. The full-core melting model merely provides a convenient way of expressing a reasonable upper limit to the magnitude of a potential nuclear-excursion energy contribution.

6. Basis of Maximum Reasonable Chemical Reaction

According to limited present knowledge, zirconium can undergo a rapid reaction with water when the metal is in a finely divided, molten state. It is not possible to rule out all chances that a small portion of the zirconium in the fuel-element cladding might under accident conditions get into a state in which it is capable of such reaction. However, the fraction of the cladding zirconium involved would be small, probably well under 25 percent. A 25-percent zirconium reaction has accordingly been used as the basis of the maximum reasonable chemical-reaction contribution toward the post-accident pressure in the enclosure.

The exact mechanism of the zirconium-water reaction is not known. However, the reaction probably involves the liberation of 2 molecules of hydrogen (H_2) per atom of zirconium reacting. An additional energy contribution due to burning of this hydrogen is conceivable. However, under the conditions that may be reasonably anticipated as potentially arising in case of a reactor-rupture accident in the Dresden Station enclosure, the hydrogen would not burn because:

a. initially it would be mixed with steam rather than with air; and

b. by the time the steam is sufficiently diluted with air to permit burning, the hydrogen concentration would be below the lower explosive limit. For example, the final hydrogen concentration in the enclosure after a 25% zirconium reaction considered under 2, above, would be 2 volume percent, while the lower explosive limit is at conditions of interest at least as high as 6%.

If hydrogen burning should nevertheless occur, it would increase the zirconium chemical-reaction contribution by 77%, i.e., a 25% zirconium reaction without hydrogen burning is equivalent to a 14% zirconium reaction with hydrogen burning.

7. Post-Accident Temperatures

Enclosure temperatures corresponding to the various post-accident pressures, determined as indicated in Appendix B, are plotted in Figure 11. On the basis of these post-accident temperatures, the design temperature rise has been conservatively set at 250°F. above ambient (i.e., at 325°F.).

8. Calculation of Post-Accident Pressures

The method employed for computing post-accident pressures in the enclosure is outlined in Appendix B.

Appendix B also lists the physical constants employed in the calculation of reasonable magnitudes of potential nuclear and chemical reaction contributions to the accident energy release.

9. Post-Accident Pressure Reduction

The post-accident pressures discussed above were calculated on the basis of instantaneous release of the pressurized hot water in the reactor system. After such a release the pressure in the enclosure would undergo changes with time as a function of two competing mechanisms:

a. There would be heat losses from the enclosure atmosphere to the enclosure shell, to solid structures in the enclosure*, and to the environment outside.

b. There would be heat gain due to radioactive decay.

Immediately after an accident, while the shell and solid masses inside are being heated up, the net effect of these

---

* While some structure portions in the sphere would be above the calculated post-accident temperature and would thus surrender heat to the enclosure atmosphere rather than withdraw heat from it, by far the largest fraction of the structure masses would be well below the post-accident temperatures, so that the net effect in the immediate aftermath of an accident would be a heat transfer from vapor space to solids.

opposing mechanisms would be to decrease the enclosure pressure.

Later radioactive decay heat would be generated faster than the enclosure could surrender heat to the environment, so that the pressure in the enclosure would rise. It is planned to provide a cooling means to supplement the natural mechanisms to insure that the pressure in the enclosure does not rise above the 37-psig test pressure during this period. (actually the cooling means contemplated is expected to have a cooling power well in excess of that needed to meet this requirement, and should reduce the post-accident pressure to 6 psig within a day.)

Eventually, as the radioactive decay power decreases, the enclosure would lose heat to the environment faster than the heat input rate, so that the internal pressure would decrease even without artificial cooling.

D. WEATHER, EARTHQUAKE, AND OTHER ALLOWANCES

The enclosure will be designed to withstand a maximum wind velocity of 110 m.p.h. (The maximum wind velocities recorded for the area are 87 m.p.h. at Chicago and 75 at Peoria. However, there has been an unofficial report of a 110-m.p.h. wind at Joliet on April 3, 1956.)

The seismographic classification of the plant site is Zone 1, with no past history of destructive earthquakes [3]. Accordingly the enclosure is being designed for a horizontal acceleration of 3.3% of gravity (slightly in excess of the 2.5% required by the Uniform Building Code, Pacific Coast Building Officials Conference, 1955 edition).

Other stresses, e.g., snow loading, and dead and live loads, are being allowed for in the design of the sphere according to the A.S.M.E. Boiler and Pressure Vessel Code.

The design allows for any combinations of weather and dead loads and internal pressure to occur simultaneously. But the maximum design live and earthquake loads are not considered simultaneous with the maximum internal pressure, because of negligible likelihood of co-occurrence.

There is no appreciable danger of flood at the contemplated plant site and accordingly no flood provisions are incorporated in the design of the enclosure. The elevation of the site is 516 feet above datum. An extreme flood elevation of 507 feet above datum at the Dresden Dam has been estimated. Thus, the highest estimated water level was still 9 feet below the site level.

The possibility of flooding on the site is very remote since spillway capacity is provided well in excess of estimated flood conditions at the Dresden Dam. Another factor which makes flooding at this site very unlikely is the fact that the elevation of the site is well above the vast valley storage area upstream from the dam.

For weather, seismological, and hydrological data for the site reference is made to document NPG-126[3].

E. LEAKAGE

The enclosure will be built to the greatest practical degree of leak-tightness. The following tests are contemplated to aid in detecting and correcting possible leakage spots*:

---

* For complete construction and testing sequence see Section G.

a. Radiographing of all welds (during construction).

b. Soap-bubble testing of all welds and seals.

c. Determining the ability of the sphere to hold a 29.5-psig air pressure for approximately two days. This leakage rate test is contemplated after the construction of the shell is complete but before the plant is placed inside it.

d. A confirmatory test at a pressure of at least 10 psig for two days or longer, after the plant is installed and temporary openings cut in the shell for plant-installation purposes are closed.

It is expected that the quality of the workmanship as aided by the radiographing and soap-bubble tests will produce a higher degree of leak-tightness than it will be possible to demonstrate reliably in the over-all leakage-rate tests (c) and (d), due to difficulties in ascertaining the exact magnitudes of the proper temperature and humidity corrections to be applied in interpreting the results of the integrated leakage tests. However, the tests are expected to be sensitive enough to show up unequivocally any leakage sufficiently severe to be of significant safety concern in the event of a worst reasonable accident.

Calculations based on assumptions outlined in Appendix C indicate that a 0.5-percent/day initial leakage rate from the enclosure in the event of a worst reasonable accident would result in an external exposure of 16 roentgens to a person standing under the centerline of the emerging cloud at the edge of the plant site (one-half mile from the enclosure) for eight hours. (Cf. proposed maximum permissible external emergency exposure of 25 r.[4]) Corresponding calculated internal exposures (integrated from the time of ingestion for a lifetime taken as infinite) are 90 rads to the thyroid and 40 rads to the bone. (Cf. 100 rads widely accepted as maximum permissible[5].) After an 8-hour cloud passage the radiation intensity from ground contamination due to fallout would be an estimated 0.5 r/hr. These calculations are based on the cooling means provided reducing the post-accident enclosure pressure from an initial 37 psig to 6 psig in 1 day.

A means of cooling the sphere contents after an accident without thermal shock to the sphere will be provided, with a view to reducing the internal pressure and maintaining it at a low value, to reduce the rate of leakage out of the enclosure through any imperfections. The cooling means will also reduce the amount of radioactive material suspended in the enclosure atmosphere through wash-down by the condensate produced. (This may be supplemented by a direct spray-down action if the type of cooling means to be selected will involve an internal spray.)

Since details of the provisions for post-accident cooling are not involved in the design and construction of the enclosure itself, detailed description and analysis of these provisions is beyond the scope of this report.

F. LOCATION

1. General

The sphere is to be located approximately 0.6 mile south of the navigation channel in the Illinois River and 0.5 mile west of the west bank of Skinner Island in the Kankakee River, the off-site land nearest to the sphere*, as shown in Figure 1.

Its distance from the south and west land boundaries of the plant property will exceed one-half mile.

These distances will provide adequate safety at the plant boundaries from direct gamma radiation from the sphere in the event of accident. Thus, in the event of a "worst reasonable accident", the integrated direct-radiation dose to an unshielded receptor one-half mile from the sphere during the first eight hours after the accident would be approximately 4 roentgens (Cf. 25 r. proposed maximum permissible emergency exposure for whole-body irradiation[4].)

2. Post-Accident Radiation

Calculated post-accident direct gamma radiation intensities and integrated doses as a function of distance from the sphere and time after the accident are shown in Table II. The method of calculation employed in arriving at the tabulated figures is outlined in Appendix D.

G. CONSTRUCTION AND TESTING SEQUENCE

The following is the contemplated sequence of major construction steps and pre-operational tests for the enclosure. The sequence was chosen on the basis of discussions with the Bechtel Corporation, engineer-constructors for the plant, and the Chicago Bridge and Iron Company, successful bidders for the subcontract for detailed design and construction of the enclosure shell.

---

* Other than approximately 0.07 square mile of State-owned flowage land located between 0.3 and 0.5 mile from the sphere.

1. Building of shell: first the equatorial course of plates, then successive courses downward, then upward from equator. Shell supported on columns at equator.

2. Radiographing of welds. (Step 2 is carried on concurrently with Step 1. New joints are being welded while previously welded joints are being radiographed.)

3. Soap-bubble leak testing of welds and seals.

4. Structural test at 37 psig (pneumatic).

5. Leakage-rate test at approximately 29.5 psig.

6. Cutting of openings in the shell to admit construction and operating equipment.

7. Placing of concrete inside and outside the lower portion of the sphere (done nearly concurrently so that no undesirable deflections or stresses are induced in the shell).

8. Transfer of a controlled part of the weight from columns to concrete foundation.

9. Interior construction.

10. Closing of temporary openings.

11. Radiographing of new welds.

12. Soap-bubble leak testing of new welds and seals.

13. Confirmatory leakage-rate test at a pressure of at least 10 psig.

14.* Confirmatory structural test at a pressure at least equal to that calculated for the maximum reasonable accident.**

H. APPLICATION OF A.S.M.E. CODE

To insure an enclosure of the highest integrity, all provisions required for the A.S.M.E. Boiler and Pressure Vessel Code approval stamp are being incorporated in the design, fabrication, and erection of the enclosure, with special considerations as indicated below.

The pertinent code sections include the latest edition and supplements of Section II (Material Specifications), Section VIII (Unfired Pressure Vessels), and Section IX (Welding Qualifications).

Significant special considerations are as follows:

a. The steel plate used will be aluminum killed and normalized, to produce maximum low-temperature impact properties.+ This additional requirement was made to provide increased protection against brittle failure.

---

* May precede Step 13.

** Possibly as high as 37 psig, but may be several psi lower.

\+ The contemplated material of construction is SA-201 Grade B Firebox-quality steel, made to SA-300 specifications.

b. No internal-pressure relief devices will be installed.* The structural safety factors employed give the sphere a good probability of containing pressures considerably greater than that calculated for the "worst reasonable accident", so that containment could be expected even for accidents of unforeseen magnitude. Also, it was thought prudent to eliminate the safety valves to avoid the possibility of malfunction and leakage that would actually make the enclosure less reliable for potential accident pressures of magnitudes which are reasonable to postulate.

c. The vessel as a whole will not be stress-relieved because of its large size. (This is in accordance with A.S.M.E. Code Committee Special Rulings in Cases Nos. 1226 and 1226-1, which permit building containment vessels for nuclear reactor installations without stress relieving, provided that plate thickness does not exceed 1.5 inches and certain other requirements are met. The sphere will be constructed of plates ranging from 1.25 to 1.40 inches in thickness.) However, any plate segment wholly containing a penetration, nozzle, or column connection will be furnace stress-relieved after insertion of the penetration. If a large penetration inter-

---

* The Code currently requires relief devices on all pressure vessels. However, at the time of this writing the A.S.M.E. Code Committee was considering possible revision of the Code to remove this requirement for nuclear-reactor containment vessels.

sects more than one shell plate and contains seams joining metal over 1.25 inches thick other than the shell plate, that portion will be furnace stress-relieved as a unit before welding into a penetration assembly or into the shell.

d. No thickness allowance will be made for corrosion. However, the vessel will be protected from corrosive conditions. It will be primed and painted in accordance with specifications of the Steel Structure Painting Council to protect against corrosion.

e. The standard hydrostatic test at 1.5 times the nominal design pressure cannot be performed since the weight of the water would collapse the sphere. A pneumatic structural test at 1.25 times the nominal design pressure will be performed as provided for in the code as an alternative.

f. The maximum stresses from live or earthquake loads are not considered simultaneous with those produced by the maximum internal pressure, because of negligible likelihood of co-occurrence.

REFERENCES

(1) NPG - 111, "Boiling and Flashing Reactor -- Central Station Power Plant Design Study (180,000 KW Net Electric)", edited by W. H. McLean and M. N. Chiarottino, Commonwealth Edison Co., February, 1955.

(2) X - GEAP - 045, "G. E. Dual Cycle Boiling Reactor -- Preliminary Summary of Basic Reactor Characteristics Affecting Safety", Atomic Products Division, General Electric Co., Schenectady, N. Y., June 14, 1955.

(3) NPG - 126, "Environmental Factors at Proposed Site of Dresden Generating Station of Commonwealth Edison Co.", by W. M. Kiefer, M. N. Chiarottino, D. W. Geue, and A. F. Veras, June 15, 1955.

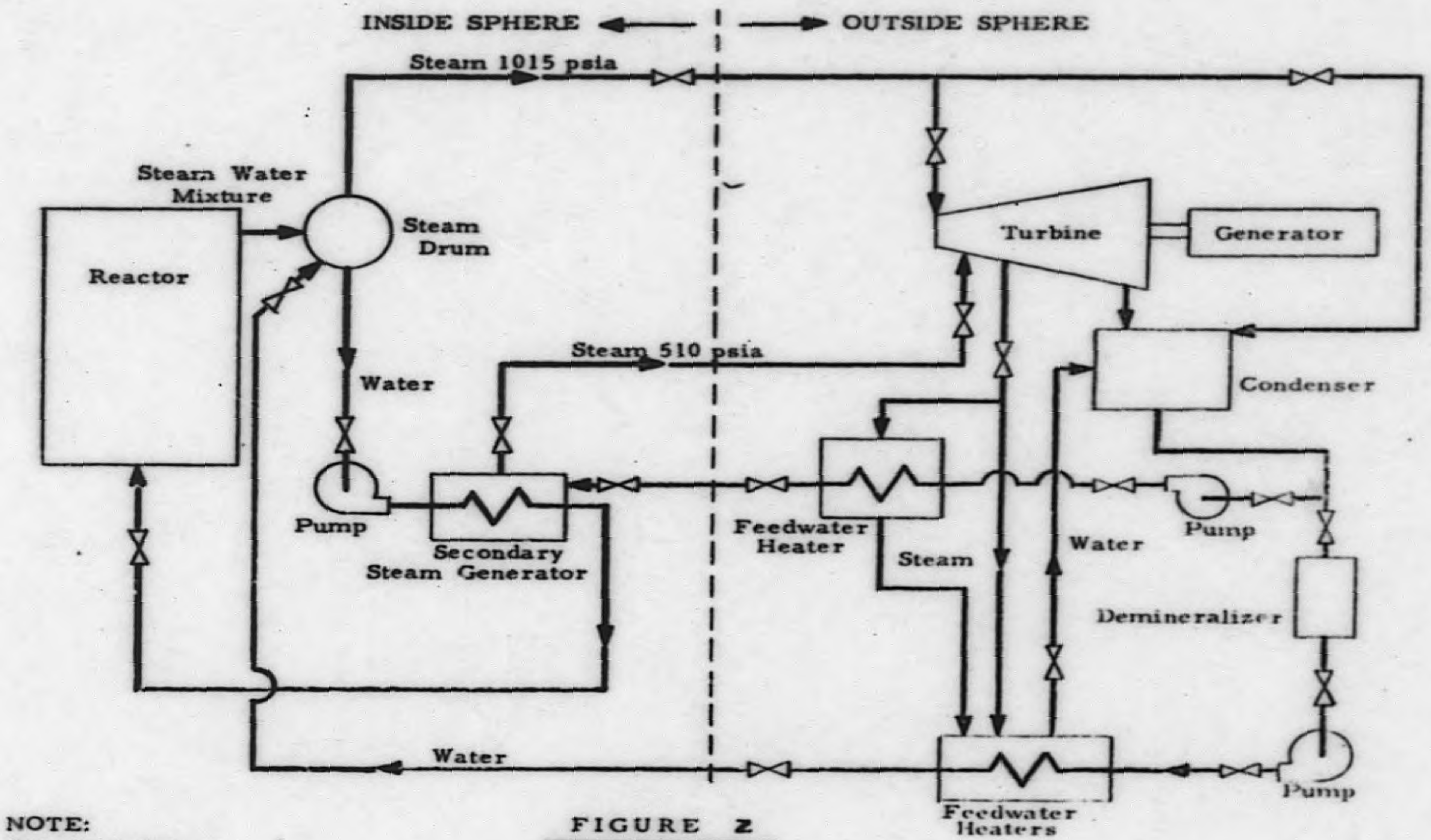
(4) "Permissible Dose from External Sources of Ionizing Radiation", U. S. National Bureau of Standards, Handbook 59, September 24, 1954. (Superintendent of Documents, Washington 25, D. C.)

(5) "Reactor Safeguards", by J. J. Fitzgerald. To be published in the "Handbook of Dangerous Materials", Reinhold Publishing Corp., New York City, New York.

(6) "Meteorology and Atomic Energy", prepared by the U. S. Department of Commerce Weather Bureau for the U. S. Atomic Energy Commission, Washington, D. C., July, 1955.

APPENDIX A

FIGURES AND TABLES



NOTE:
Principal Valves
Shown by Symbol: ⋈

FIGURE 2

FLOW DIAGRAM

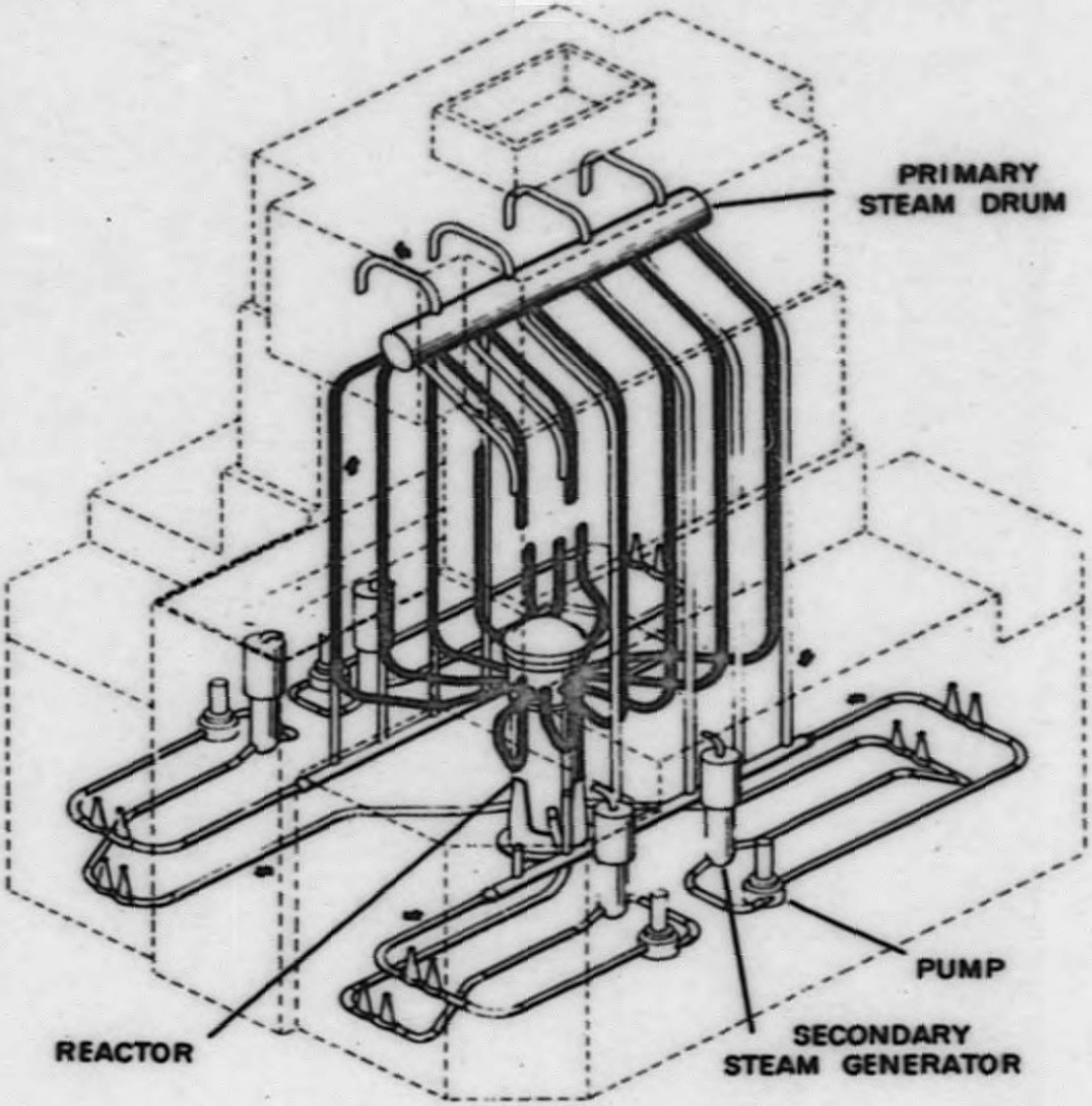


FIGURE 3
Perspective View of Nuclear Steam Generation System

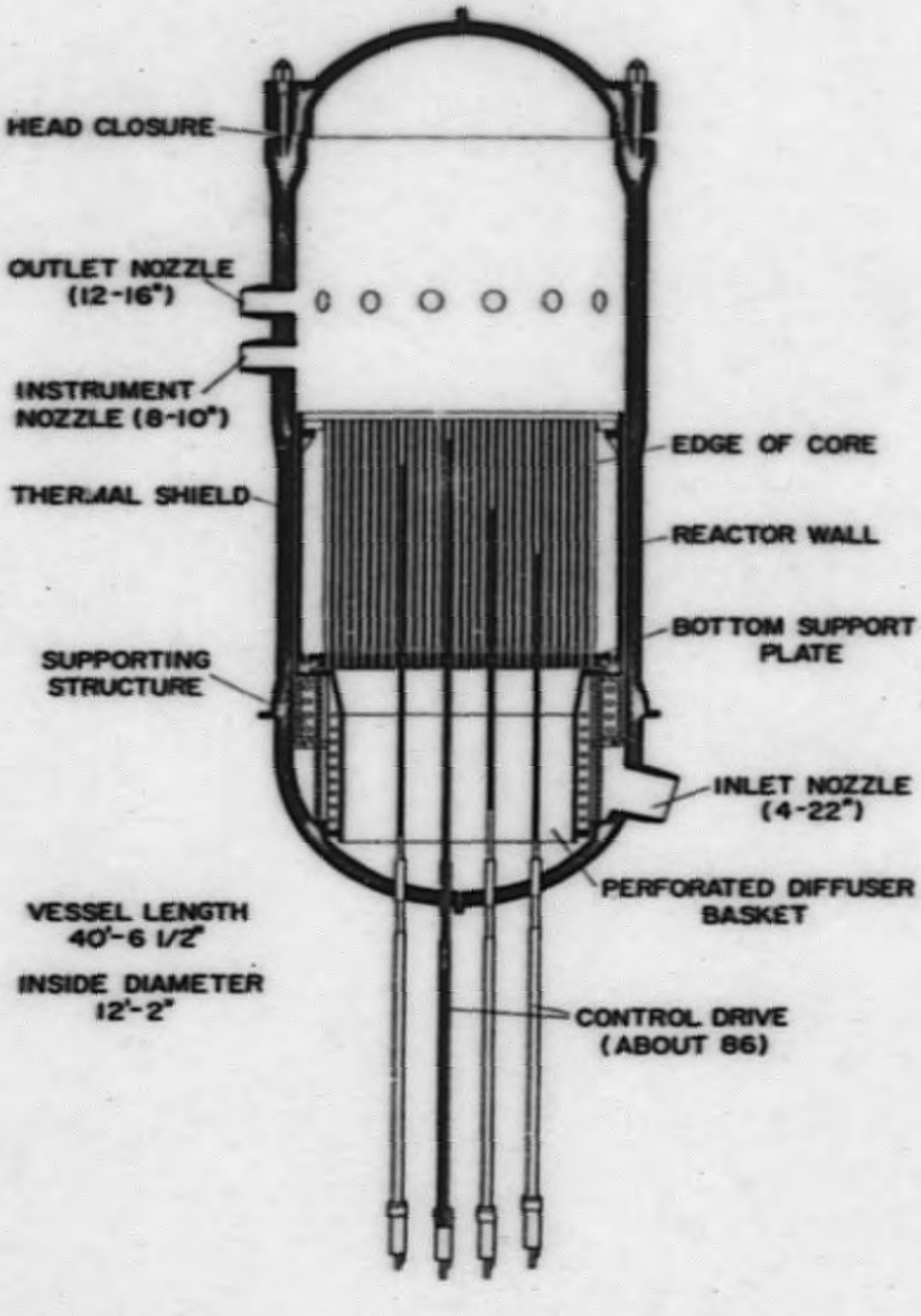


Figure 4
LONGITUDINAL SECTION OF REACTOR

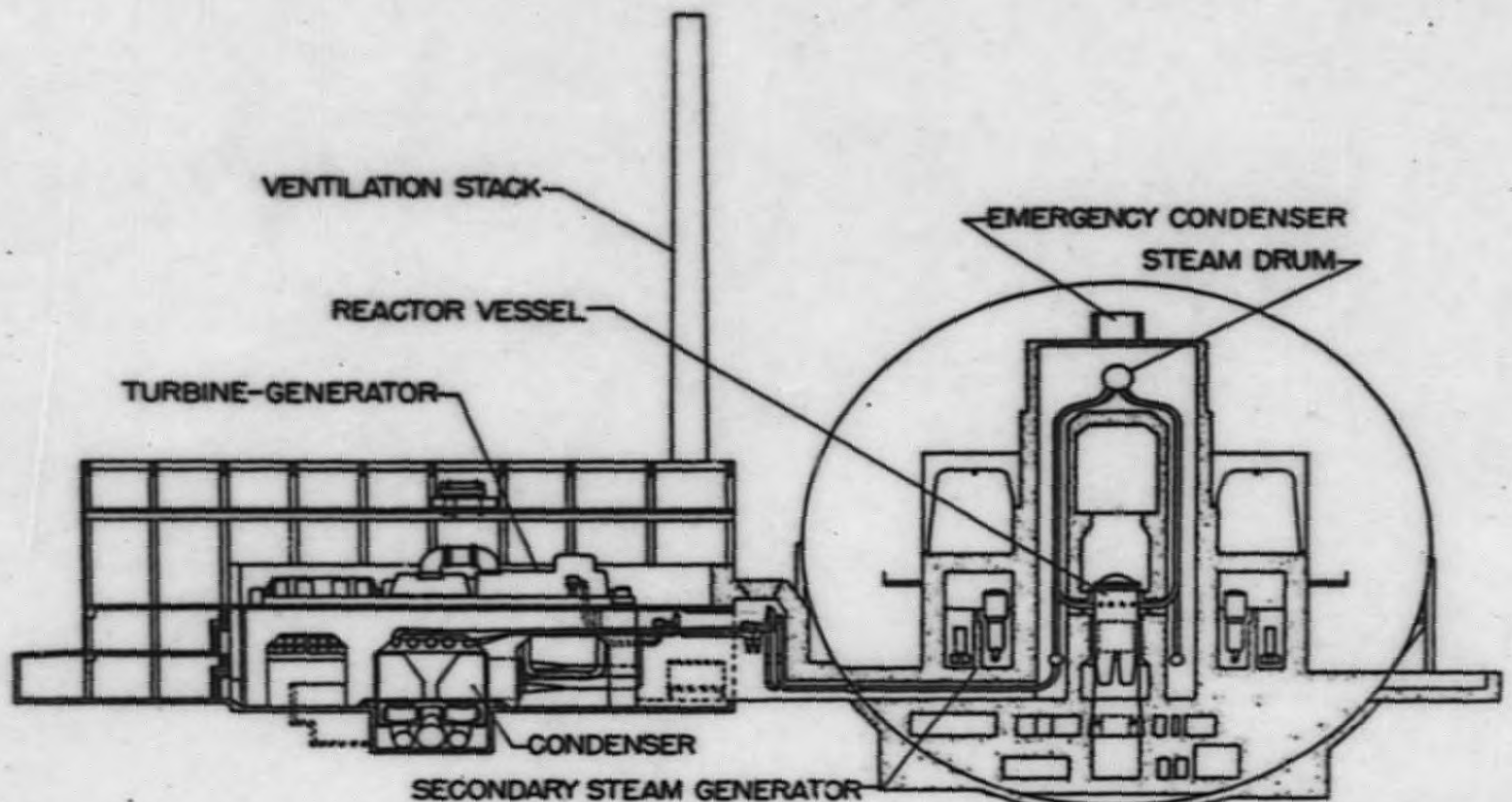
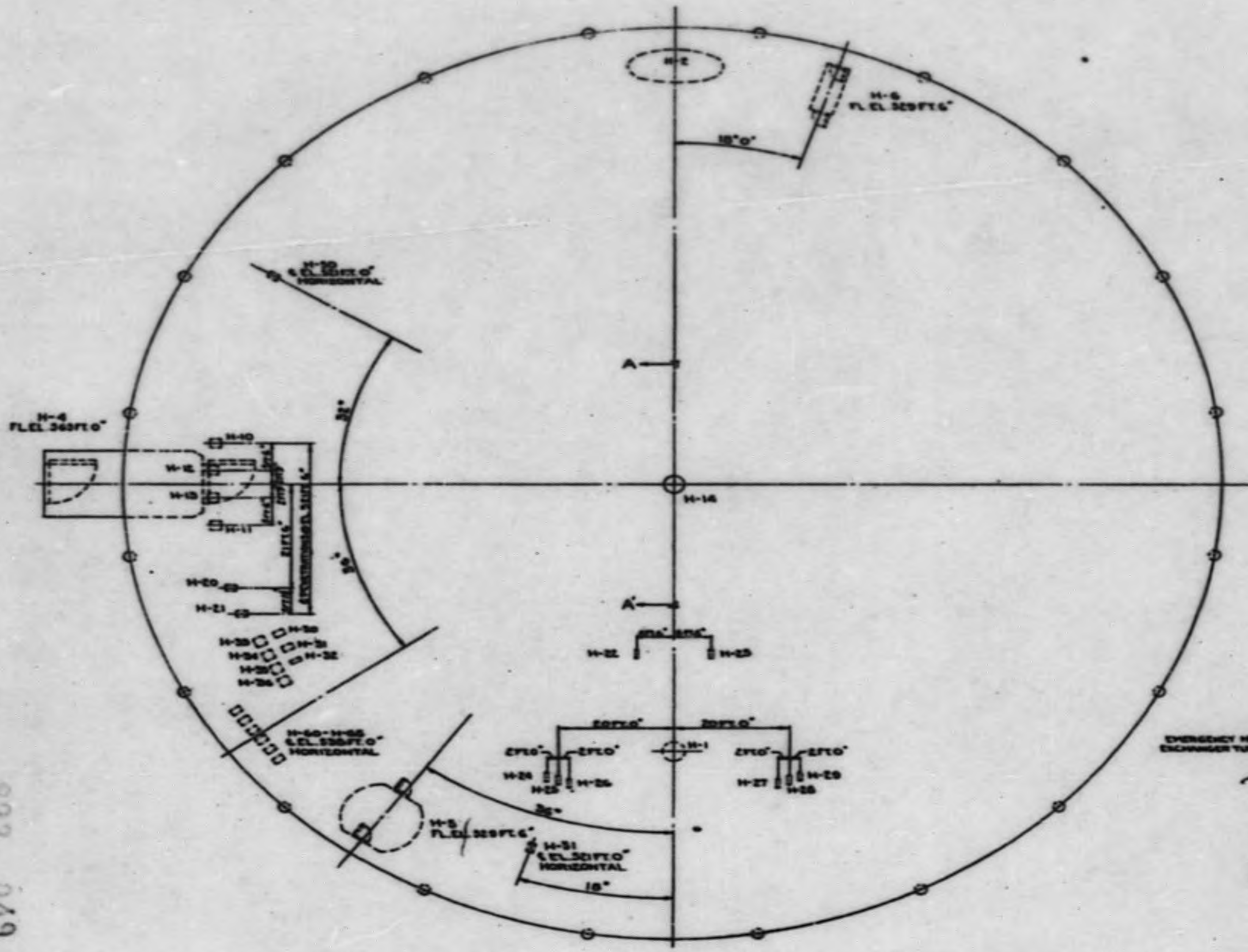
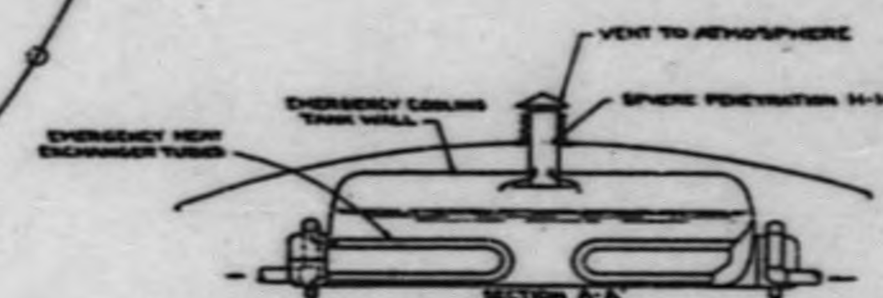


FIG. 5 PLANT EQUIPMENT ARRANGEMENT



PENETRATION SCHEDULE

NO.	SIZE	LENGTH	SERVICE
H-1	40" I.D.		FUEL HANDLING
H-2	17' DIA.		BOLTED SERVICE OPENING
H-4	5'x8' DOOR		ACCESS LOCK (FL. EL. 365'-0")
H-5	2'-6"x6'-0" DR.		ACCESS LOCK (FL. EL. 329'-6")
H-6	2'-6" DIA. DR.		ESCAPE LOCK (FL. EL. 329'-6")
H-10	28" I.D.	24"	PRIMARY STEAM
H-11	28" I.D.	24"	PRIMARY STEAM
H-12	28" I.D.	24"	SECONDARY STEAM
H-13	28" I.D.	24"	SECONDARY STEAM
H-14	42" O.D.	24"	EMERGENCY COOLING WATER TANK VENT
H-20	19" I.D.	24"	PRIMARY FEEDWATER
H-21	19" I.D.	24"	SECONDARY FEEDWATER
H-22	6"	24"	POST INCIDENT COOLING WATER
H-23	6"	24"	do.
H-24	6"	24"	do.
H-25	6"	24"	do.
H-26	6"	24"	do.
H-27	6"	24"	do.
H-28	6"	24"	do.
H-29	6"	24"	do.
H-30	14" O.D.	24"	COOLING WATER
H-31	14" O.D.	24"	COOLING WATER
H-32	8"	24"	DEMINERALIZED WATER
H-33	24" O.D.	24"	MISC. PIPING
H-34	24" O.D.	24"	do.
H-35	24" O.D.	24"	do.
H-36	24" O.D.	24"	do.
H-50	16" O.D.	24"	VENTILATION EXHAUST
H-51	16" O.D.	24"	VENTILATION INLET
H-60	12"	24"	PNEUMATIC INSTRUMENTATION
H-61	12"	24"	do.
H-62	12"	24"	do.
H-63	12"	24"	do.
H-64	12"	24"	do.
H-65	12"	24"	do.



THIS IS NOT FUNCTIONALLY A PENETRATION THROUGH THE ENCLOSURE WALL. FUNCTIONALLY THE ENCLOSURE ENDS AT THE TANK AND TUBE WALLS AS SHOWN.

FIG. 9 - SPHERE PENETRATION LAYOUT - PLAN

INFORMATION FROM A DWG.
PREPARED BY BECHTEL CORP.

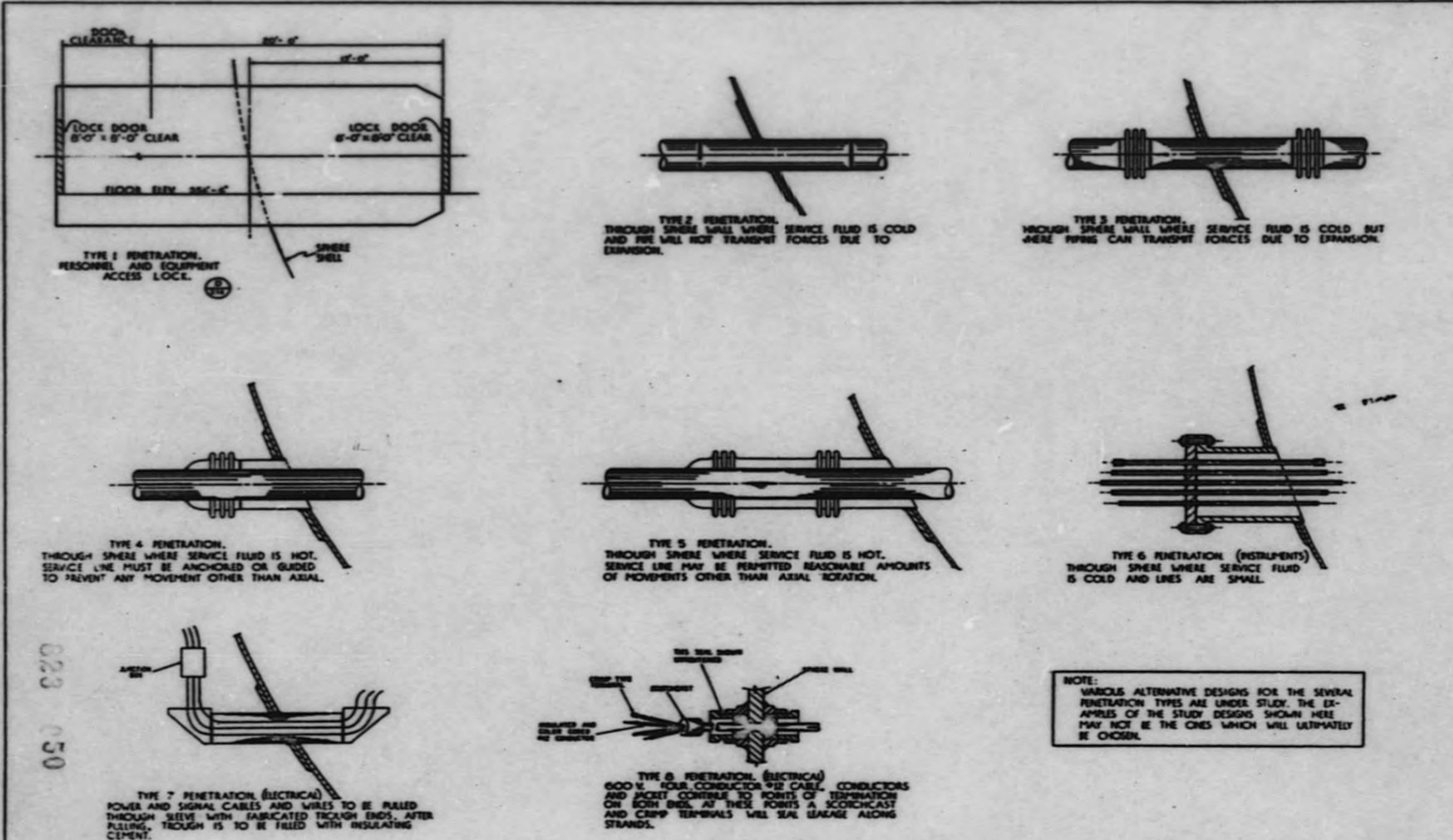


FIG. 10 - PENETRATION TYPES

190-Foot-Diameter Sphere; 80% Free Volume

Remarks:

1. Water Tonnages:

Loop	Tons of Water at 1000 PSIG, Sat. Temp.	
	Normal Op.	Highest Internal Energy*
Primary	188	212
Primary and Secondary	203**	237

* Up to pressure, but not boiling.

** Adjusted for lower temperature in secondary loop. (This adjusted weight -- the proper one to use with the plot below -- is 3 tons lower than the actual water weight.)

2. Maximum reasonable nuclear-excursion energy (energy to melt all UO_2 in reactor core): 48 million B.t.u.

3. Example chemical-reaction (Zr-H_2O reaction) energies:

% of Zr in Fuel Cladding Reacting	Energy, Millions of B.T.U.	
	With Liberated H_2 Not Burning	With H_2 Burning
10	9	16
25	23*	40

* Considered maximum reasonable chemical energy contribution.

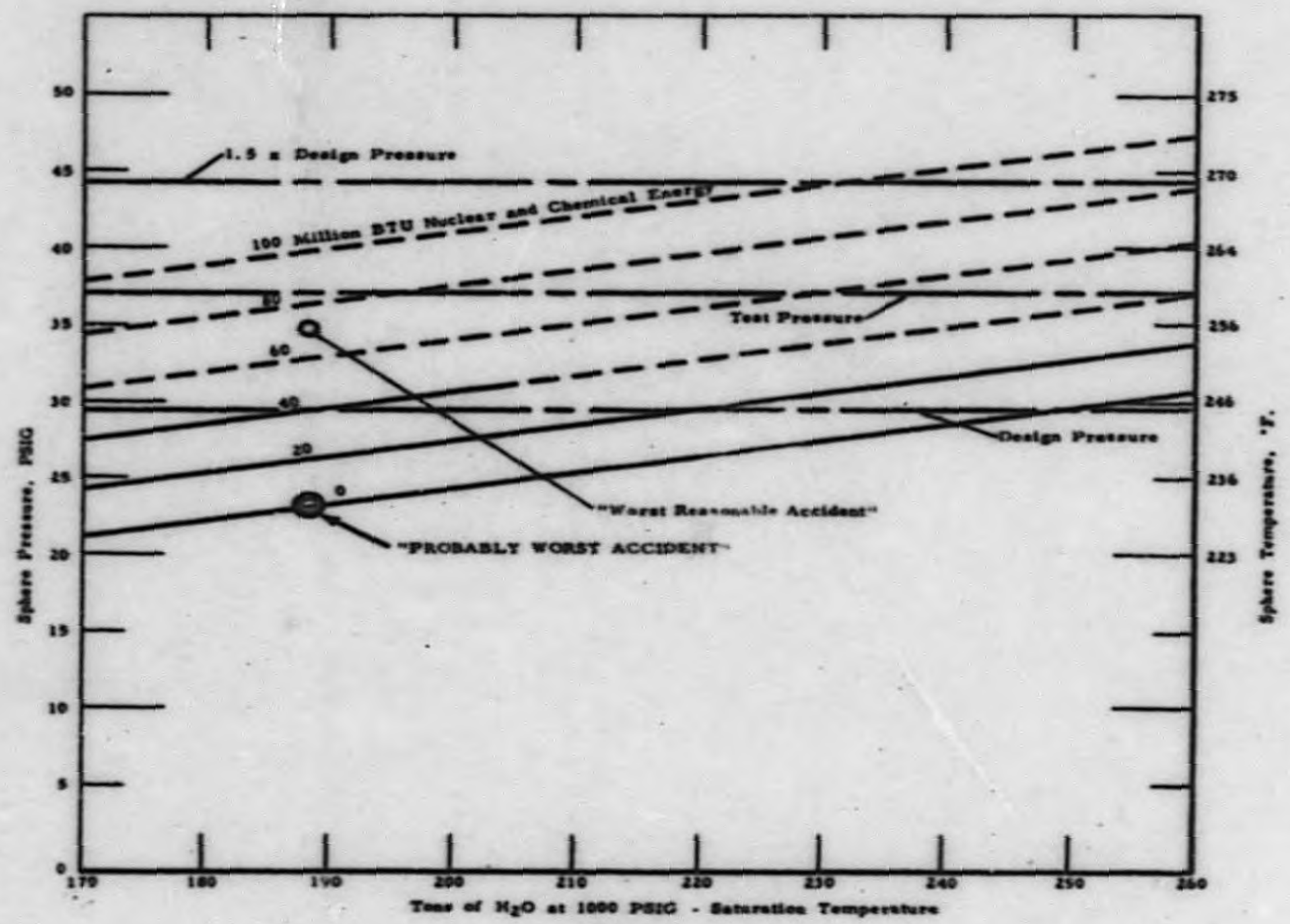


FIGURE 11

POST - ACCIDENT PRESSURES

TABLE I

ENCLOSURE PRESSURES AFTER VARIOUS POTENTIAL ACCIDENTS

Note: Design pressure for enclosure: 29.5 PSIG

Enclosure test pressure: 37 PSIG

Case	Pre-Accident Conditions	Accident	Tons of Pressurized Water Liberated	Nuclear and Chemical Energy Contribution, Millions of B.T.U.			Enclosure Pressure, PSIG
				Nuclear	Chemical	Total	
1	Fuel change: reactor head open; reactor water cool. ("Young" fuel)	Coolant loss.	0	0	0	0	<1
2**	Normal power operation	Reactor or pipeline rupture with coolant loss.	188	0	0	0	23**
3	Normal power operation	Reactor rupture + nuclear excursion with enough energy to melt 1/4 of the UO_2 in reactor core.	188	12	0	12	25
4	Highest water internal energy (theoretical extreme: reactor up to 1000-psig pressure, but not boiling)	Reactor or pipeline rupture with coolant loss.	212	0	0	0	26
5	Normal power operation	Case 3 + chemical reaction of 10% of the zirconium in the fuel cladding with water.	188	12	9	21	27
6	Normal power operation	Reactor rupture + secondary steam generator rupture + "1/4-core" nuclear excursion + 25% Zr reaction	203*	12	23	35	30
7	Normal power operation	Reactor rupture + nuclear excursion with enough energy to melt all of the UO_2 in the core	188	48	0	48	31
8	Normal power operation	Case 7 + 10% Zr reaction.	188	48	9	57	32
9	Normal power operation	Case 7 + 10% zr reaction + hydrogen burning.	188	48	16	64	34
10***	Normal power operation	Case 7 + 25% Zr reaction.	188	48	23	71	35***
11§	Normal power operation	Case 10 + secondary steam generator rupture	203*	48	23	71	36§
12§	Normal power operation	Case 10 + H_2 burning.	188	48	40	88	[illegible]§
13§	Normal power operation	Case 11 + hydrogen burning.	203*	48	40	88	[illegible]§
14§	Normal power operation	Reactor rupture + secondary steam generator rupture + 2-core nuclear excursion + 50% Zr reaction + H_2 burning.	203*	96	80	176	55§

* Adjusted for lower temperature in secondary loop. (This adjusted weight is 3 tons lower than the actual water weight.)

** "Probably worst accident" (i.e., accident creating a pressure which probably would not be exceeded in any accident).

*** "Worst reasonable accident" (i.e., worst accident to which it is reasonable to assign a possibility of occurrence.

§ Hypothetical-accident case with no significant possibility of occurrence, presented only to aid in evaluating the degree of conservativeness represented by the structural safety factors

TABLE II

POST - ACCIDENT RADIATION

Dosage Rates vs. Distance and Time

Distance from Sphere Center	Roentgens/Hour			
	After 1 Hr.	After 1 Day	After 1 Mo.	After 1 Year
600 feet	200	70	7	0.5
1/2 mile	0.7	0.2	0.02	0.002
1 mile	0.001	0.0003	<0.0001	<0.0001

Integrated Dose vs. Distance and Time

Distance from Sphere Center	Total Roentgens		
	First 5 Minutes	First Hour	First 8 Hours
600 feet	200	500	1000
1/2 mile	0.6	2	4
1 mile	0.001	0.003	0.007

APPENDIX B

CALCULATION OF POST-ACCIDENT PRESSURE

Outline of Method

The post-accident pressure developed inside the sphere (as a result of the liberation of the internal energy of saturated boiling water depressurized from 1000 psig to near-atmospheric pressure and the liberation of energy from a nuclear excursion and/or chemical reaction between reactor components) was calculated by the solution of three equations involving three unknowns: post-accident pressure, post-accident temperature, and energy released. The relationship between ultimate temperatures and pressures was found for selected post-accident temperatures from the sum of the partial pressure of air (as determined from the ideal-gas laws) and of the partial pressure of steam (as determined from steam tables).

The amount of steam formed, or the fraction of water evaporating, was then determined from the known container volume (80% free volume in a 190-ft.-diameter sphere plus the volume of the reactor vessel) and the specific volume of steam at the calculated post-accident partial steam pressure.

The energy from the nuclear excursion and chemical reaction could then be found by subtracting the original internal energy of the reactor water plus the internal energy of the pre-accident enclosure atmosphere from the internal energy of the fraction of reactor water remaining after the accident and the internal energy of the post-accident enclosure atmosphere.

The original enclosure atmosphere was taken as air, saturated with water vapor, at 100° F. Other assumptions necessary included adherence to the gas laws, perfect air-steam mixing, and no energy loss by absorption in structural materials, by radiation through the sphere or loss of energy from neutron and gamma radiation during the nuclear excursion.

The results of these calculations allowed preparation of curves relating post-accident pressures and temperatures in a 190-foot-diameter sphere to energy release for various tonnages of liberated pressurized hot water. (See Figure 11, in Appendix A.)

Physical Constants

The following are values of physical data employed in the computation of the nuclear and chemical energy contributions:

Weight of uranium oxide (UO_2) in reactor core	126,000 lb.
Specific heat of UO_2	0.08 B.t.u./lb.
Latent heat of fusion of UO_2	107 B.t.u./lb.
Melting point of UO_2	4360° F.
Average UO_2 temperature during operation	1000° F.
Weight of zirconium (Zr) in fuel cladding (including that in screw connectors between fuel segments)	30,600 lb.
Heat of reaction (exothermic) between Zr and water	151 Kcal./g.mol.* (2970 B.t.u./lb. Zr)*
Heat of reaction for hydrogen burning	(2280 B.t.u./lb. Zr)

* Based on assumed formation of crystalline ZrO_2 in the reaction. On the basis of the formation of fused ZrO_2 the heat of reaction would be about 1/3 of this value (840 B.t.u./lb. Zr).

A P P E N D I X   C

CALCULATION OF EFFECTS OF LEAKAGE

The post-accident, off-site personnel exposure to radiation emanating from a cloud of fission products that might occur after a serious nuclear accident were calculated by postulating a small imperfection in the sphere that resulted in a leakage rate of 15.3 m^3/hr. (0.5% of the sphere free volume per day) at an initial post-accident pressure of 37 psig. As the sphere atmosphere cools (by heat loss to the concrete inside the sphere, by heat loss through the sphere walls, and by use of the post-accident cooling system which will be designed to remove enough heat to bring the sphere pressure to 6 psig or less in 24 hours) the pressure will be reduced and the leakage rate decreased. A curve of leakage rate vs. internal pressure was prepared, on the basis of the standard orifice flow equation.

From the pressure - vs. - leakage curve and the predicted cooling rate, a second curve of leakage rate vs. time was drawn. Integration under this curve gave an estimate of the total leakage expected for any period of time and thus an estimate of the amount of sphere atmosphere released to the environs.

The resulting integrated dosages to a stationary observer, 1/2 mile distant from the sphere, were then calculated assuming continued dispersal of 30% of the fission products during the post-accident period and using photon-energy and exposure data from reference (5). Standard formulas for inversion meteorological conditions [(6)] were applied to obtain cloud concentrations at 1/2 mile. It was further postulated that fission-product radioactivity build-up in the fuel had essentially reached equilibrium.

Radiation levels, resulting from ground deposition were calculated on the basis of maximum fall-out at each position, corrected for decay and total leakage for the periods of interest.

No corrections were made for removal of the fission products in the sphere atmosphere by falling condensate (although a significant removal will undoubtedly occur) or for the observer not staying under the centerline of the cloud during the entire periods of interest. The lack of these corrections should insure pessimistic values for this study.

APPENDIX D

CALCULATION OF POST-ACCIDENT RADIATION

Post-accident radiation levels were calculated at selected distances from the sphere following an accident involving a nuclear excursion of four seconds' duration and developing an energy of 48 million B.t.u. It was postulated that 30% of the solid fission products (built up during the burst and during a 365-day reactor operation period) and all of the halogen and noble-gas fission products were dispersed to the sphere atmosphere and thus were "visible". One hundred percent dispersal of the gaseous fission products was assumed (to allow for possible "cook out" of the fission products by the heat developed in the uncooled fuel rods.) Account was taken of attenuation through air, the steel shell of the sphere (taken as 1.25 inches thick) and water contained in the sphere atmosphere, but not of self-shielding of the fuel or possible shielding by other structural components.

Corrections were applied for radioactive decay plus particulate and halogen removal by wash-out which would reduce radiation levels as these processes occurred. Wash-out rates for particulates were determined from data given in "Meteorology and Atomic Energy"(6) for assumed formation of 10-micron particles and a "rainfall" inside the sphere of 0.05 in./hr. Halogen removal rates were obtained from the same source, on the basis of all halogens being absorbed by droplets of condensing steam to the same extent as I_2 vapors. The "rainfall rate" of 0.05 in./hr. was based on the amount of water condensed as energy in the sphere atmosphere was removed.

APPENDIX E

MISSILE SHIELDING

With a concrete radiation shield around the sides of the reactor certainly massive enough to stop any fragments formed in a reactor rupture accident under any reasonably conceivable conditions, a possible need for special missile shielding provisions exists only for:

(a) primary missiles (i.e., fragments of the reactor itself) propelled up or down, and

(b) in certain places, secondary missiles (i.e., for bits of structure surrounding the reactor).

Extensive studies by General Electric and by Broadview Research and Development under contract to G. E. are being devoted to these questions:

(a) Is there enough missile hazard to warrant missile shielding?

(b) If so, what constitutes an adequate measure of protection?

Answers to these questions are not now at hand. However, for study purposes, missile shielding is being designed on the basis of arbitrarily assumed safe-side answers to the above questions. Further study and evaluation are in progress in an effort to determine the actual requirements.

The present study design of the missile shielding consists in the following components:

1. For primary missiles:

   (a) Reactor vessel cover shield.

   This is an approximately 225-ton steel-plate and concrete structure, which can be wheeled out of the way for fuel change. (Analytical work is now under way on the incorporation of a wooden "cushion" in the design of this shield.)

   (b) Extra reinforcing steel in the reinforced-concrete tunnel above the reactor.

   (c) Two 12-inch-thick steel slabs (with 6-1/2 feet of concrete in between) near the sphere bottom, below the reactor.

2. For secondary missiles:

   (a) Woven-cable blast mats at the ends of the tunnel.

   (b) A small steel or concrete shield to protect the refueling tube.

Provisions 1(a), 1(c), and 2(a) are depicted on Figure 7.

This missile shielding design is based on controlling a theoretical accident in which the reactor lid, weighing 50 tons, is torn off and propelled upward in an explosive reactor-rupture accident, involving 20 million B.t.u. of explosion energy. (This is about three times the explosive energy that would be liberated in a zirconium-water reaction involving 25% of the zirconium in the fuel cladding if such a reaction liberated as high a fraction of explosive energy as a TNT explosion.)

Other types of hypothesized severe reactor-rupture accidents, involving reasonably conceivable combinations of conditions, are also being analyzed and taken into consideration in the missile-protection studies. The design of missile shielding provisions depends on assumptions as to pressure in the reactor at the instant preceding rupture, the time course of the rupture-causing event and of the rupture, the maximum missile weight, the shape of the missile and its orientation in flight, and the properties of the materials involved that determine the reaction upon impact between missile and shield.

Turbine explosion missile protection is not needed at the Dresden Station: the turbine is outside the enclosure and rotates in a plane not intersecting it.

Primary coolant system valves are invariably so located that at least approximately 4 feet of concrete are interposed between valve and enclosure wall,* eliminating any potential danger to the enclosure from valve fragments.

---

* With the exception of certain small valves (8 in. and smaller) subject to primary-system pressure. The circumstances of these are being analyzed to determine what restrictions on design, location, or orientation are prudent from the missile-potential standpoint.

APPENDIX F

POWER SUPPLY

The sources of electrical power supply to plant auxiliary power system, including the interior of the enclosure, are as follows:

A. Normal Power Sources

1. Normal running power:

   Approximately equally divided between auxiliary transformer connected to generator terminals and auxiliary transformer connected to the 138-kv transmission system of Commonwealth Edison Company. Either transformer can supply full-load auxiliary power.

2. Normal shutdown (also Plant Start-up) power:

   From one full-capacity auxiliary transformer connected to 138-kv Commonwealth Edison Company system.

B. Emergency Power Sources

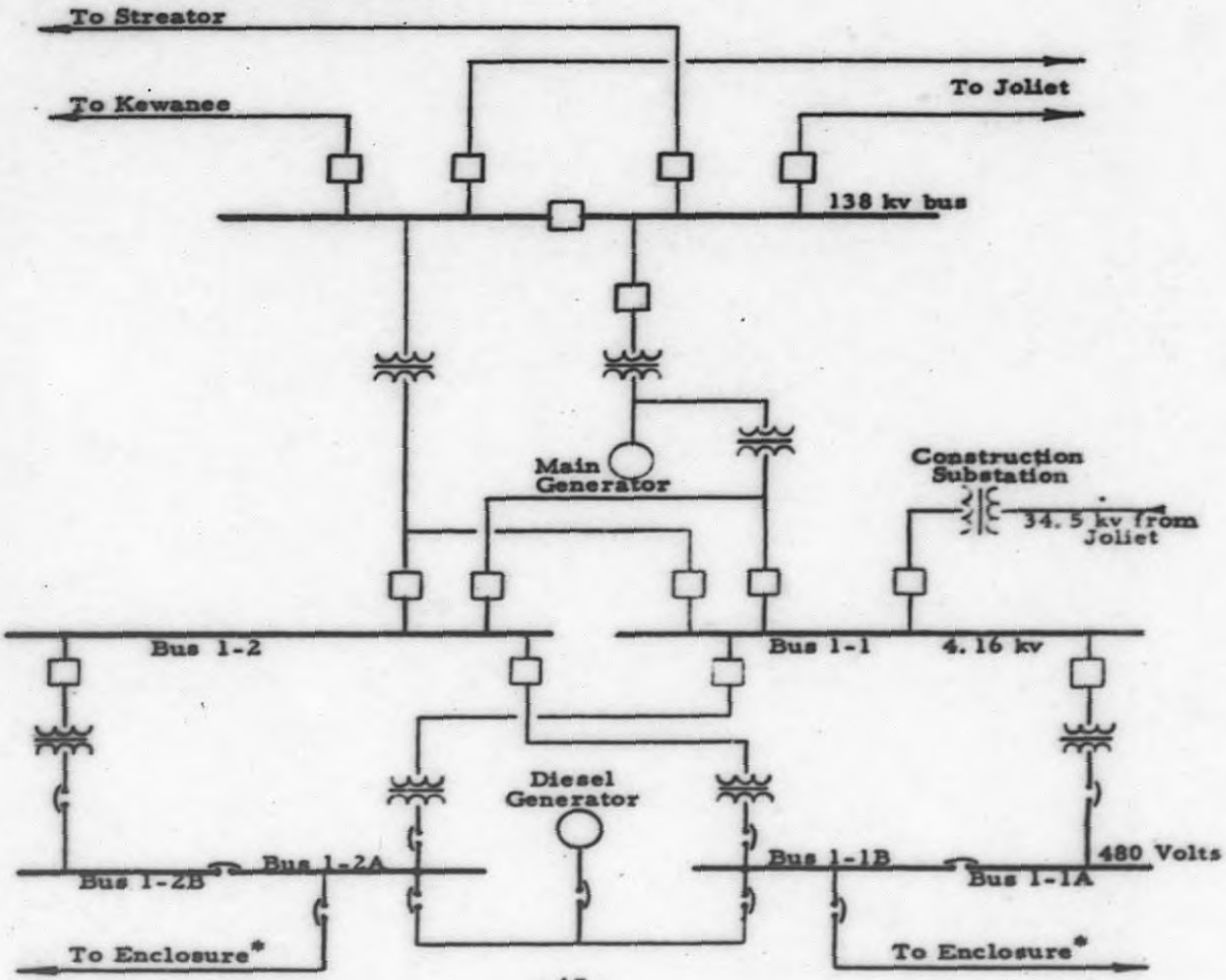
Either of the following independent power sources is capable of supplying the power needed during shutdown or during post-accident period when the normal shutdown power source is not available:

1. 2500-kva substation connected to the 34.5-kv distribution system of Commonwealth Edison

Company. (This is the substation used to supply construction power and is larger than actually needed for emergency power.)

2. Approximately 250 kw available from a diesel generator located in the turbine building.

C. <u>One - Line Diagram:</u>



* Separate circuits to all critical equipment.

523 062

A P P E N D I X   G

PLANT PARAMETERS

(For preliminary orientation only: Subject to modification in course of detailed design.)

Power Output

Reactor power	626,000 kw
Gross electrical output	192,000 kw
Net electrical output	180,000 kw

Fluid Flow

Coolant flow, forced circulation	25,600,000 lbs/hr.
Inlet temperature	505°F

Nuclear Core

Diameter (nominal)	10' 6"
Height (active)	8'10"
Total UO_2 in core	123,672 lbs.
Sub-assemblies in core, total	712
Fuel rods in core, total	17,044
Zirconium, structural (flow channels)	8,600 lbs.
Zirconium in clad and fuel-segment connectors	30,600 lbs.
$\frac{\text{Water volume}}{UO_2 \text{ volume}}$	2.1 : 1

Fuel

Type	Compacted, sintered UO_2, zirconium clad rods
Density, UO_2	90% of theoretical
UO_2 dimensions in fuel rod	0.50 inch x 106 inches
Average enrichment	1.4% in U^{235}
Clad thickness (zirconium)	0.03 inches
Fuel-rod spacing, center to center	0.74 inches

(Plant Parameters, continued)

Control Rods

Total number	84
Composition	40% cadmium, 60% silver
Normal velocity	6 inches per second
Maximum velocity (insertion only)	9 feet per second
Rod strength (average)	0.17% Δ k
Withdrawal rate	0.01% Δ k per second
Total rod strength	14.5% Δ k

Heat Transfer

Average heat flux - at rated power	98,000 Btu/hr./sq.ft.
Maximum heat flux - at rated power	300,000 Btu/hr./sq.ft.
Maximum power per unit fuel-rod length	12.9 kw/ft.

Reactor

Inside diameter	12' 2"
Height (over-all)	42 feet
Operating pressure	1015 psia
Shell - ASTM A-302 steel	5-1/4 inches
Clad - AISI 304 ELC steel	3/8 inch

Turbine

Type	Tandem compound double-flow
Rating	192,000 kw at 2.5" mercury absolute, 1800 rpm on saturated steam

Generator

Capacity - 245,000 kva at 85% power factor. 14,400 volt, hydrogen cooled.	Three-phase, 60 cycle

(Plant Parameters, continued)

Main Condenser

Type	Single-flow, divided-box
Size	Approximately 120,000 sq.ft.

Primary Steam Drum

Size	8' 6" I.D. x 60 ft.

Secondary Steam Generators

Number	4
Type	Vertical "U"-tube

Feedwater Heaters

Number	10, with 5 extraction points
Type	Vertical "U"-tube

Pumps (Centrifugal)

System	Number Required for Normal Operation	Number of Installed Spares	Capacity (Each), GPM	HP Rating
Recirculation	4	0	17,500	650
Primary feedwater	2	1	1,650	1750
Condensate	2	1	3,100	350
Secondary feedwater	2	1	1,400	800
Condenser circulation	2	0	90,000	500