DRAFT
8-29-66

DOCKET NUMBER
PROD. & UTIL. FAC. 50-249

UNITED STATES

ATOMIC ENERGY COMMISSION

In the Matter of the Application of

COMMONWEALTH EDISON COMPANY

Formal

Docket No. 50-249

APPLICANT'S EXHIBIT NO. 1

SUMMARY DESCRIPTION OF DRESDEN UNIT 3 AND EVALUATION OF CONSIDERATIONS IMPORTANT TO SAFETY

1.0 INTRODUCTION

1.1 General

Subsequent to the issuance of Provisional Construction Permit No. CPPR-18 on January 10, 1966 in AEC Dkt. No. 50-237 authorizing construction of Unit 2 at Dresden Nuclear Power Station, Commonwealth Edison Company determined, subject to the receipt of the necessary authorizations, to proceed with the construction of Unit 3 adjacent to Unit 2 (see Plate 1, Station Perspective). It was also determined that Dresden Units 2 and 3 would be identical in virtually all respects, e.g. design concepts and criteria, capacity and components, and that any improvements incorporated in Unit 3 would also be incorporated in Unit 2. Accordingly, the only differences contemplated in the design of Units 2 and 3 relate solely to the location of equipment and, for desired diversity, the use by Unit 3 of the 345 KV system as a source of normal auxiliary power, rather than the 138 KV system source to be used for Unit 2.

The design of Unit 3, however, differs from the Unit 2 design which was reviewed by the AEC in Dkt. No. 50-237 prior to issuance of CPPR-18* because of (i) changes in the interconnections of certain auxiliary systems and sharing certain common facilities, (ii) changes related to providing a turbine-generator for Unit 3 which is approximately 2% larger in capacity than originally considered for Unit 2, (iii) changes developed in finalizing the design of Units 2 and 3, and (iv) refinements in certain analyses.

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*All changes in the Unit 2 design required to conform with the Unit 3 design will be submitted for review by amendments of the Unit 2 Plant Design and Analysis Report filed in AEC Dkt. No. 50-237.

Each of the differences in the Unit 3 design from that previously reviewed for Unit 2 are identified and analyzed in the Unit 3 Plant Design and Analysis Report as amended filed in Docket No. 50-249. Plate 2, attached hereto, provides an index to each section of the Unit 3 Report in which a change in design is described or analyzed. None of such changes involve any changes in the Principal Architectural and Engineering Criteria, set forth in section 4.0 hereof, which govern the development of the design of both Units 2 and 3.

A detailed evaluation has been made of the manner in which the Unit 3 design satisfies the "General Design Criteria for Nuclear Power Plants" published by the Atomic Energy Commission on November 22, 1965 and utilized by the AEC staff for evaluation of Unit 2 in AEC Kt. No. 50-237. Such evaluation is set forth in Appendix H of the Unit 3 Plant Design and Analysis Report.

Since Appendix H was filed as a part of Amendment No. 1 to the Unit 3 Plant Design and Analysis Report, several design changes have been effected by subsequent amendments, principally Amendment No. 5. Specifically, such changes provide (i) in lieu of one of the two isolation condenser systems originally proposed, a high pressure coolant injection system which, in addition to serving as a backup for the remaining isolation condenser system, can restore any loss of coolant resulting from postulated breaks of limited size in the primary system, (ii) additional redundancy in components of each of the core spray systems, (iii) a low pressure coolant injection system capable of promptly reflooding the core to prevent significant core melting after a postulated accident resulting in the loss of coolant at the maximum credible rate, (iv) redundancy in emergency power, (v) emergency coolant supply, (vi) automatic relief valve operation, (vii) two separate and independent containment spray systems for each unit 2 and 3, and (viii) redundant components in the rod worth minimizer.

Each of the foregoing changes provides additional assurance that the Commission's General Design Criteria will be satisfied and that Unit 3 can be constructed and operated without jeopardizing public health and safety.

1.2 Applicability to Statement of Issues in Notice of Hearing

This document is responsive principally to the first issue, and incidentally to the fourth issue, stated in the Notice of Hearing issued herein dated August 22, 1966. With respect to sub-paragraph (1) of the first issue, the following sections 2.0, 3.0, 4.0, 5.0, and 6.0 provide a description of the proposed design or the facility, set forth the principal architectural and engineering criteria for design and identify the major features of components on which technical information is required. Section 6.0 also commits the Applicant to furnish such technical information and, therefore, is responsive to subparagraph (2) of the first issue.

In relation to the issues raised in subparagraph (3) of the first issue, the Applicant considers that there are no unresolved safety questions respecting Unit 3 and, accordingly, that a research and development program is not required to resolve safety questions. Even though further tests and design effort are required to confirm analyses and establish final detailed design of certain components, these are matters which lie beyond the issues relevant at this stage of the licensing process. As was stated in the Initial Decision dated December 29, 1965 entered in AEC Dkt. 50-237, "all such efforts will be analyzed and evaluated by the Applicant and the Staff as the project goes forward and all will be considered prior to licensing the facility for operation."

While the document in its entirety is responsive to subparagraph (4) (other than the clause respecting unresolved safety questions) of the first issue, the Applicant considers that sections 2.0, 7.0 and 8.0 are particularly relevant in this respect.

2.0 SITE AND ENVIRONS

The site for Dresden Unit 3 is the Dresden Nuclear Power Station owned by Commonwealth and consisting of approximately 953 acres in Goose Lake Township, Grundy County, Illinois. In addition, Commonwealth leases from the State of Illinois approximately 17 acres lying in two narrow strips of river frontage along the northeast and east edges of the Station. The Station boundaries generally follow the Illinois River to the north, the Kankakee River to the east, a country road from Devine extended eastward to the Kankakee River on the south and the Elgin, Joliet and Eastern Railroad right-of-way on the west.

The Dresden Station serves as the site for Dresden Unit 1, a General Electric boiling water reactor, which is and has been in operation since 1960. Drésden Unit 2, located immediately west of Unit 1, is under construction pursuant to CPPR-18 issued January 10, 1966 pursuant to the Initial Decision entered in AEC Dkt. 50-237. Dresden Unit 3 is to be located immediately west of and adjacent to Unit 2. The eastermost unit, Unit 1, will be situated approximately 0.5 mile from the center line of the Kankakee River to the east and the westernmost unit, Unit 3, will be located approximately one mile from the west boundary of the site and all of the units will be situated approximately 0.5 mile south of the center line of the navigational channel in the Illinois River and 0.5 mile north of the south boundary of the site (see Plates 3 and 4).

There are no residences on the site. Portions of the site outside the fenced area occupied by Unit 1 and required in the construction of Units 2 and 3 are leased to a neighboring farmer for cattle grazing and production of field crops.

Dresden Nuclear Power Station was thoroughly investigated as a site for a nuclear power reactor and found to be suitable by the Atomic Energy Commission in 1956 when the construction permit for Dresden Unit 1 (CPPR-2) was issued. The successful operation of Unit 1 since 1960 and the environs monitoring programs carried on by Commonwealth, the State of Illinois and Argonne National Laboratory have confirmed such findings.

The suitability of the site was re-examined in 1965 during the course of proceedings in AEC Dkt. 50-237. Studies in the areas of meteorology, geology, seismology, hydrology, population density and land usage in the environs were prepared and submitted for review by the Atomic Energy Commission and other governmental agencies, including the Office of Meteorological Research, Geological Survey, Fish and Wildlife Service of the U. S. Department of the Interior and the Coast and Geodetic Survey of the U. S. Department of Commerce and seismologic consultants employed by the Atomic Energy Commission, as well as the Atomic Safety and Licensing Board in Docket 50-237. On the basis of such studies and review it was concluded by all of the foregoing agencies that the Dresden Station provides a suitable site for construction and operation of Unit 2. (See Initial Decision dated December 29, 1965, entered in AEC Dkt. 50-237).

The same studies and reviews equally support the conclusion that Dresden Station is a suitable site for construction and operation of Unit 3 and satisfy the criteria set forth in 10 CFR 100. Details of the site evaluation studies and supporting data are set forth in Volume III of the Unit 2 Plant Design and Analysis Report submitted in AEC Dkt. 50-237 and are summarized in Sections I-2 and I-3 of the Unit 3 Plant Design and Analysis Report.

3.0 SUMMARY DESCRIPTION OF THE FACILITY

3.1 Introduction

Dresden Unit 3 is to be a General Electric boiling water reactor identical in its design features to Dresden Unit 2, the construction of which was recently authorized in the proceedings on the application of the Commonwealth Edison Company in Atomic Energy Commission Docket 50-237. Certain common facilities are provided for Units 1, 2 and 3 which are summarized in Section 5 hereof.

Almost all of the features of Unit 3 have been demonstrated successfully in the operation of one or more of the General Electric boiling water reactors which have been constructed, including the basic fuel design, zircaloy fuel cladding, hydraulically operated control rods, in-core neutron monitoring instrumentation, pressure suppression containment and radioactive waste control. All new features (see Section 6.0 hereof) which have not yet been utilized on an operating reactor will have been demonstrated on Dresden Unit 2, which will be placed into operation approximately one year prior to Unit 3.

A technical description of Unit 3 is given in Section I-4.0 of the Plant Design and Analysis Report. The following is intended to highlight the principal design features which are significant to safety considerations, and Table 1 summarizes plant design features and some data appropriate to achieve a reactor thermal output of 2255 MW. While some of the parameters listed in the table may be refined as design and procurement progresses, the final design will satisfy the principal architectural and engineering criteria for design as set forth in Section 4 hereof. A drawing of the reactor and containment typical of that contemplated for Unit 3 is shown in Plate 5.

3.2 Reactor Primary System

The reactor is a single cycle, forced circulation, boiling water reactor producing steam for direct use in the steam turbine.

The fuel for the reactor core consists of slightly enriched uranium dioxide pellets contained in sealed Zircaloy-2 tubes. These fuel rods are assembled into individual fuel assemblies of 49 fuel rods each. A complete core loading consists of 724 fuel assemblies.

Control of the core is achieved by 177 movable control rods and 324 temporary control curtains. The control rods consist of assemblies of 3/16-inch diameter, sealed, stainless steel tubes containing compacted boron powder and held in a cruciform array by a stainless steel sheath fitted with castings at each end. The control rods are of the bottom-entry type and are moved vertically within the core by individual, hydraulically operated, locking piston type control rod drives. The control rod is withdrawn or inserted at a limited rate, one rod at a time, for power level control and flux shaping during reactor operation. Stored energy available from gas-charged accumulators and from reactor pressure provides hydraulic power for rapid simultaneous insertion of all control rods for reactor shutdown. Each drive has its own separate control and scram device. The temporary control curtains fabricated of boron stainless steel will be fixed between fuel channels during the early life of the initial core to supplement the control rods. Reactor power level control is augmented by controlling the recirculation flow rate through the reactor core.

A standby liquid control system is provided as an independent redundant control mechanism to be used in the remote event that the control rod system become inoperative.

The reactor pressure vessel contains the reactor core and structure, the steam separators and dryers, the jet pumps, the control rod guide tubes, the feedwater, core spray and standby liquid control spargers, and other components. The main connections to the reactor vessel include the steam lines, jet pump recirculation lines, feedwater lines, control rod drive thimbles and other connections for core cooling. Plate 6 is a cut-away view of the reactor vessel and core arrangement.

Reactor coolant enters the bottom of the core and flows upward through the fuel assemblies where boiling produces steam. The steam water mixture is separated by steam separators and dryers located within the upper portion of the reactor vessel, and the steam passes through steam lines to the turbine. The separated water mixes with the incoming feedwater within the reactor vessel and is returned to the core inlet through jet pumps located within the reactor vessel. The motive force for the jet pumps is supplied by the water from the two reactor coolant recirculation loops. The recirculation pump motors receive electrical power from variable frequency motor-generator sets which are used to vary pump speed and the resultant recirculation flow rate as a means of reactor power level control.

3.3 Containment Systems

Unit 3 employs an independent primary containment system identical to that for Dresden Unit 2. The primary containment, consisting of a drywell, a pressure-suppression chamber and interconnecting vent pipes, provides the principal containment barrier surrounding the reactor pressure vessel and recirculation cooling system. Any leakage from the primary containment is to the secondary containment system which consists of the reactor building, the standby gas treatment system, and the 310 foot stack. These latter features of the secondary containment are common to both Units 2 and 3.

The primary containment is designed to accommodate the pressures and temperatures which would result from or occur subsequent to a failure equivalent to a circumferential rupture of a recirculation line within the primary containment. The pressure suppression chamber is a steel torus-shaped pressure vessel approximately half filled with water, and located below and encircling the drywell. The vent system from the drywell terminates below the water level of the pressure suppression chamber so that in the event of a pipe failure in the drywell the released steam would pass directly to the water where it would be condensed. This transfer of energy to the water pool would reduce rapidly (within 30 seconds) the residual pressure in the drywell and substantially reduce the potential for subsequent leakage from the primary containment.

Two features are included in the primary containment design to aid in maintaining the integrity of the primary containment system indefinitely in the event of a loss-of-coolant accident. First, two independent, redundant cooling systems are included for the removal of heat within the drywell and the pressure suppression chamber. Second, provision is made in the containment structure design to inert or control the composition of the containment atmosphere during plant operation. The inert atmosphere is intended to preclude a hydrogen-oxygen reaction should any appreciable metal-water reaction occur subsequent to a loss-of-coolant accident.

The primary safeguards functions of the secondary containment are to minimize ground level release of airborne radioactive materials and to provide for controlled, filtered, and elevated release of the building atmosphere under accident conditions. The reactor building which encompasses the primary containment of

Units 2 and 3 provides secondary containment when the primary containment is in service, and primary containment during periods when the primary containment is open. For these reasons, the reactor building is designed as a controlled leakage structure. A standby gas treatment system is provided to treat the reactor building atmosphere and exhaust it to the 310 foot stack during containment isolation conditions.

The safeguards features of both the primary containment and the reactor building are capable of being tested periodically.

3.4 Auxiliary and Standby Cooling Systems

In addition to the turbine generator and main condenser system, multiple independent auxiliary systems are provided for reactor and containment cooling under various normal and abnormal conditions.

(a) A shutdown cooling system is provided which circulates water from the reactor through heat exchangers and back to the reactor for the removal of reactor decay heat.

(b) An isolation condenser system is provided for dissipation of decay heat when the reactor is under full pressure and isolated from the main condenser. The isolation condenser has its own contained cooling water with replenishment from two independent sources.

(c) A high pressure coolant injection system is provided for dissipation of decay heat as backup to the isolation condenser, and to provide coolant inventory control and heat dissipation to the suppression chamber under postulated slow depressurization accidents.

(d) Two core spray systems are designed to pump water under accident conditions from the pressure suppression chamber pool directly to the reactor core by a spray header or sparger mounted in the reactor vessel above the core.

(e) A low pressure coolant injection system is provided to pump water from the suppression chamber pool through heat exchangers directly to the inner vessel surrounding the core, thus reflooding the core and maintaining cooling of the fuel assemblies subsequent to postulated loss of coolant accident.

(f) An emergency coolant supply system, provided by a cross-tie between the service water system and the feedwater system, makes available an inexhaustible supply of cooling water to the core and containment independent of all other cooling water sources.

(f) In addition, the design provides for two independent primary containment spray cooling systems which pump water under accident conditions from the pressure suppression chamber pool through a heat exchanger to spray nozzles discharging into the drywell.

3.5 Plant Control and Instrumentation

Reactor power is controlled by movement of control rods and by regulation of the recirculation flow rate. Control rods are used to bring the reactor through the full range of power and to shape the core power distribution. Load following and adjustments in reactor power level are accomplished with recirculation flow control. Procedural controls supplemented by protective devices are used so that thermal performance does not exceed established limits. Also, to minimize the consequences of postulated reactivity insertion, a rod worth minimizer is provided to restrict control rod patterns to those in which the worth of any rod does not exceed a nominal $0.025\Delta k$.

A bypass system having a capacity of approximately 40% of steam flow at rated load is supplied with the turbine to restrict overpressure transients resulting from sudden turbine control valve or stop valve closure. Partial load rejection can be accommodated by the rapid action of the bypass system.

The reactor protection system overrides the above control systems to initiate any required safety action. The reactor protection system automatically initiates appropriate action whenever the plant conditions monitored by the system approach pre-established limits. The reactor protection system acts to shutdown the reactor, close isolation valves, or initiate the operation of other safeguards systems as required. Components of the reactor protection system can be removed from service for testing and maintenance without interrupting plant operations and without negating the ability of the protection system to perform its protective functions upon receipt of the appropriate signals.

Instrumentation is provided for continuous monitoring of the radioactivity of specific process systems and to provide alarms or signals for appropriate corrective actions. In addition, radiation surveillance of the site and environs will be maintained.

Controls necessary for safe operation of the reactor are located in the Unit 3 control room in the turbine building.

3.6 Fuel Storage and Handling

New fuel is stored in a dry vault in the reactor building. Refueling is conducted under water, and the irradiated fuel is stored under water in the reactor building until shipment from the site.

3.7 Electrical System

The electrical output of Unit 3 will feed into the 345 KV network through five 345 KV circuits. The primary feature of the electric power system serving Dresden Station is the diversity of dependable power sources, physically isolated so that any one instrument of failure affecting one source of supply will not communicate to alternate sources, thus assuring a continuous source of auxiliary power to Unit 3. Auxiliary power can be supplied from five separate and independent sources: Unit 2, Unit 3, the 345 KV transmission system, the standby diesel generator system and a 34.5 KV line. Three standby diesel generators are provided, any two of which can serve the emergency requirements of one unit and the normal shutdown requirements of the other.

A station battery for Unit 3 is used for all controls vital to plant safety and to power certain functions required for a safe shutdown, such as closing of isolation valves, opening valves to the isolation cooling system, providing lighting, and required instrumentation. Examples of such required instrumentation are the control rod position indicators and a neutron channel to monitor the core during shutdown.

3.8 Radioactive Waste Control

Gaseous, liquid and solid waste control facilities are provided to limit the release of radioactive materials from the site in accordance with applicable regulations.

TABLE 1

PRINCIPAL DESIGN FEATURES OF DRESDEN UNIT 3

Station	
Net Electrical Output	715 MW
Reactor	
Thermal Output	2255 MW
Core Operating Pressure	1000 psig
Total Core Flow Rate	98 x 10^6 lb/hr
Steam Flow Rate	8.62 x 10^6 lb/hr
Core	
Circumscribed Core Diameter	189.7 inches
Active Length	12 feet
Fuel Assembly	
Number of Fuel Assemblies	724
Fuel Rod Array	7 x 7
Cladding Material	Zircaloy-2
Fuel Material	UO_2
Active Fuel Length	144 inches
Cladding Outside Diameter	0.570 inch
Cladding Thickness	0.036 inch
Fuel Channel Material	Zircaloy-4
Control System	
Number of Movable Control Rods	177
Shape of Movable Control Rods	Cruciform
Pitch of Movable Control Rods	12.0 inches
Control Material in Movable Control Rods	Boron Carbide
Type of Control Drives	Bottom entry, hydraulic actuated
Material in Temporary Control Curtains	Boron - Stainless steel
Number of Temporary Control Curtains	324
Control of Reactor Power Output	Movement of control rods and variation of coolant flow rate
Core Design Data and Operating Conditions	
Power Density	36.7 kw/liter
Heat Transfer Surface Area	63,527 sq. ft.
Average Heat Flux	116,300 Btu/hr-ft^2
Maximum Heat Flux	349,000 Btu/hr-ft^2

Core Design Data and Operating Conditions (Cont'd)

Minimum Critical Heat Flux Ratio at Overpower - Equal to or Greater Than	1.5
Core Average Voids of Coolant within Assembly	37%
Core Average Exit Quality of Coolant within Assemblies	9.9%

Design Power Peaking Factors

Total Peaking Factor	3.0
Additional Allowance for Overpower	1.2

Nuclear Design Data

Initial Average Fuel Enrichment	2.0%
Water/UO_2 Volume Ratio	2.38
Excess Reactivity of Clean Core (Uncontrolled) at 63°F.	0.26Δk
Total Worth of Control	0.30Δk
Reactivity of Core with All Control Rods In	0.96 k
Worth of Standby Liquid Control System	0.17Δk

Reactor Vessel

Inside Diameter	20 ft. - 11 in.
Overall Length	68 ft. - 7 5/8 in.
Design Pressure	1250 psig

Coolant Recirculation Loops

Location of Recirculation Loops	Containment Drywell
Number of Recirculation Loops	2
Pipe Size	28 inches
Number of Jet Pumps	20
Location of Jet Pumps	Inside Reactor Vessel

Primary Containment

Type	Pressure Suppression
Design Pressure of Drywell Vessel	62 psig
Design pressure of Suppression Chamber Vessel	62 psig
Maximum Leakage Rate	Less than 0.5% Free Volume per Day

Secondary Containment

Type	Reinforced concrete and steel superstructure with metal siding
Internal Design Pressure	0.25 psig

Secondary Containment (Cont'd)

Maximum Inleakage Rate	100% free volume per day at 0.25 in. water negative pressure

Structural Design

Seismic Resistance	0.1g
Sustained Wind Loading	110 mph
Control Room Shielding	Dose not to exceed 500 mrem in 8 hours under accident conditions assuming full core melt

Unit Electrical Systems

Number of Incoming Power Sources	5 - 345 KV 1 - 34.5 KV
Separate Power Sources Provided	2 Auxiliary transformers 3 Standby diesel generators shared between Units 2 and 3 1 Standby transformer 1 Station battery

Reactor Instrumentation System

Location of Neutron Monitor System	In-core
Ranges of Nuclear Instrumentation	
Startup Range	Source to 0.01% rated power
Intermediate Range	0.0001% to 10% rated power
Power Range	1% to 125% rated power

Reactor Protection System

Number of Channels in Reactor Protection System	2
Number of Channels Required to Scram or Effect Other Protective Functions	2
Number of Sensors per Monitored Variable in Each Channel	2
Method to Prevent Unwarranted Withdrawal of Control Rods	Automatic interlocks

Other Engineered Safeguards - Summary of Systems and Functions

2 Core Spray Systems	To cool the core under assumed loss of coolant accident; capability to reflood the core following coolant loss.
Low Pressure Coolant Injection System	To provide a means of cooling the core by reflooding the core subsequent to a postulated loss of coolant accident.

Other Engineered Safeguards - Summary of Systems and Functions (Cont'd)

2 Containment Spray Cooling Systems	To remove energy from primary containment subsequent to assumed loss of coolant accident.
High Pressure Coolant Injection System	To provide backup cooling capacity to the isolation condenser and to provide high pressure coolant makeup for postulated slow depressurization accidents.
Emergency Coolant Supply	To provide an inexhaustible supply of water to the core and containment independent of all other core cooling methods.
Rod Velocity Limiter	To limit the free fall of a control rod from the core to approximately five feet per second.
Control Rod Drive Thimble Support	To prevent a control rod drive mechanism from falling away from the reactor pressure vessel in the unlikely event of a failure of a drive thimble.
Main Steam Line Flow Restrictors	A constriction in each main steam line to reduce rate of blow-down in event of postulated severance of the main steam line.
Isolation Condenser	To avoid overheating of the reactor fuel in the event that reactor feedwater capability is lost and other normal heat removal systems which require a-c electrical power for operation are not available.
Isolated Valves	To effect reactor containment automatically when required under postulated accident conditions.
Primary Containment Inerting System	To provide inert atmosphere in the primary containment system to preclude a hydrogen-oxygen reaction subsequent to a postulated coolant loss accident.

Other Engineered Safeguards - Summary of Systems and Functions (Cont'd)

Standby Gas Treatment System	To provide a means for removal of particulates and halogens from Units 2 and 3 reactor building air under postulated accident conditions prior to discharge of the filtered air through the Units 2 and 3 stack. Also provides a means for maintaining the reactor building at a negative pressure so that leakage is into the reactor building and thus prevents ground level release of building air under postulated accident conditions.
Standby Liquid Control System	To provide a redundant, independent back-up control mechanism in the event that the control system becomes inoperable.

4.0 PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA FOR DESIGN

The principal architectural and engineering criteria for design for the plant are summarized below.

4.1 Plant Design

Principal structures and equipment which may serve either to prevent accidents or to mitigate their consequences will be designed, fabricated and erected in accordance with applicable codes and to withstand the most severe earthquakes, flooding conditions, windstorms, ice conditions, temperature and other deleterious natural phenomena anticipated at the site during the lifetime of this unit.

4.2 Containment

a. The primary containment, including the drywell, pressure suppression chamber, associated access openings and penetrations, will be designed, fabricated and erected to accommodate, without failure, the pressures and temperatures resulting from or subsequent to the double-ended rupture or equivalent failure of any coolant pipe within the drywell.

b. Provisions will be made both for the removal of heat from within the primary containment and for such other measures as may be necessary to maintain the integrity of the containment system as long as necessary following a loss of coolant accident.

c. The reactor building, encompassing the primary containment system, will provide containment when that system is open and secondary containment when the primary containment system is closed.

d. Provision will be made for initial preoperational pressure and leak rate testing of the entire primary containment system and for leak rate testing at periodic intervals after the facility has commenced operation. Provision will also be made for demonstrating the functional integrity of reactor building containment.

e. The integrity of the complete containment system and such other engineered safeguards as may be necessary will be designed and maintained so that off-site doses resulting from postulated accidents will be below the values stated in 10 CFR 100.

4.3 Reactor System

a. A direct-cycle boiling water reactor will be employed to produce steam at 1000 psig for use in a steam-driven turbine-generator. The reference design thermal output of the reactor is approximately 2255 MWt.

b. The reactor will be fueled with slightly enriched uranium dioxide contained in zircaloy clad fuel rods.

c. The minimum critical heat flux ratio and maximum fuel center temperature evaluated at the design overpower condition will be below values which could lead to rod failures.

d. Fuel rod cladding thickness will be designed to maintain cladding integrity throughout the anticipated fuel life. Fission gas release within the rods and other factors affecting design life must be considered for the maximum expected exposures.

e. The reactor and plant will be designed so that there will be no inherent tendency for undamped oscillations.

f. The reactor will be designed to accommodate tripping of the turbine-generator, loss of power to the reactor recirculation system and other station transients and maneuvers which might be expected without compromising safety and without fuel damage.

g. The reactor will include separate systems including overpressure scram, the isolation condenser, safety valves, and turbine bypass, to prevent serious primary reactor system overpressure.

h. Power excusions which could result from any credible reactivity addition accident will not cause damage, either by motion or rupture, to the pressure vessel or impair operation of required safeguards.

i. Heat removal systems will be provided which are capable of safely accommodating core decay heat under all credible circumstances, including isolation from the main condenser and loss of coolant from the reactor. Each different system so provided will have appropriate redundant features.

j. Reactivity shutdown capability shall be provided to make and hold the core adequately subcritical by control rod action, from any point in the operating cycle and at any temperature down to room temperature, assuming that any one control rod is fully withdrawn and unavailable for use.

k. Redundant backup reactivity shutdown capability will be provided independent of normal reactivity control provisions. This system will have the capability, with adequate margin, to shut down the reactor from any operating condition.

4.4 <u>Control and Instrumentation</u>

a. The plant will be provided with a centralized control room having adequate shielding to permit occupancy during all design accident situations.

b. There will be sufficient interlocks or other protective devices so that procedural controls are not the only means of preventing

c. A reliable reactor protection system will be provided to automatically initiate appropriate action whenever plant conditions approach pre-established limits. Periodic testing capability will be provided. Sufficient redundancy will be provided so that failure or removal from service of any one component or portion of the system will not preclude scram or actuation of other protective devices when required.

4.5 Electrical Power

Sufficient normal and emergency auxiliary sources of electrical power will be provided to assure a capability for prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances.

4.6 Radioactive Waste Control

a. Gaseous, liquid and solid waste control facilities will be designed so that discharge of effluents and off-site shipments shall be in accordance with 10 CFR 20.

b. Process and discharge streams will be appropriately monitored and such automatic features incorporated as may be necessary to prevent releases above the permissible limits of 10 CFR 20.

4.7 Shielding and Access Control

The radiation shielding in the unit and the station access control patterns will be such that the doses shall not exceed those specified in 10 CFR 20.

4.8 Fuel Handling and Storage

Appropriate fuel handling and storage facilities will be provided to preclude accidental criticality and to provide cooling for spent fuel.

5.0 INTERACTIONS OF UNITS 1, 2 AND 3

As noted above in Section 3.0, certain portions of the Station are utilized in common or are shared wholly or partially by the three units. The criteria followed for the design of this multi-unit Station with regard to the sharing of components, systems, and facilities is that each unit will operate independently of each other. A malfunction of equipment or an operator error in any of the units will not affect the continued operation of the remaining units. Likewise, if by the sharing of a component, system, or facility, the safety of the plant is increased, then such sharing will be accomplished.

There are a number of systems and facilities which are shared by all three units but which are not safety related. These include such facilities as the Administration Building, Access Control Building, Machine Shop, Laundry, Sewage Treatment Plant, and Plant Security Fencing. Certain service facilities are also shared including the Fire Protection System, Circulating Water System, Service Water System, Make-up Demineralizer System, Service Air System, and Heating System. Failure of such systems would not jeopardize safety of the units.

Following is a summary of other shared systems and facilities to which special attention has been directed to assure adequate levels of safety during normal operation of the units and postulated accident conditions.

a. Site and Off-Site Environs Monitoring

The present environs monitoring programs will be continued with multi-unit operation. The purpose of the program is to provide adequate monitoring for the integrated Dresden Station complex and to assure that the requirements of 10 CFR 20 are met for the com-

b. New Fuel Storage

Unit 1 will continue to utilize its own new fuel vault, whereas the storage vault for Units 2 and 3 has been specifically designed for use as a common new fuel storage facility.

c. Reactor building Closed Cooling Water System

This system serves as an interface between certain other systems which are exposed to the reactor and the river water, and prevents potential leakage of radioactive materials directly to the river. Sharing of these systems provides greater flexibility of plant operations which is assessed as contributing to overall plant safety.

d. Auxiliary Power System

The three units are tied together electrically only through the Edison transmission system. Unit 3 derives its auxiliary power from the 345 KV system and Unit 2 from the 138 KV system. An electrical fault in one auxiliary electrical system will not propagate to the other auxiliary systems.

e. Standby Diesel Generator System

The standby diesel generator system includes three diesel generators, any two of which are capable of furnishing power requirements for operation of emergency equipment on one unit and normal shutdown power for the other.

f. Station Battery

Each unit has a separate station battery. However, a 100 percent capacity spare battery charger is provided which can be connected to the d-c system of either Unit 2 or Unit 3. Electrical breakers are interlocked to prevent connecting the d-c systems of both Units together.

g. Liquid and Solid Radioactive Waste Control

The radwaste facility for Units 2 and 3 is designed for use as a common radioactive chemical treatment plant. The purpose of the plant is to process liquid radioactive wastes for reuse; to discharge treated wastes to the river, and to concentrate certain wastes for storage and off-site shipment.

The Unit 1 radwaste facility is entirely separate from the facility for Units 2 and 3. Both facilities are designed to maintain the concentrations of radioactive wastes released to the environment within the limits of 10 CFR 20. The aggregated discharges of liquid wastes will remain within limits now applicable solely to Unit 1.

h. Gaseous Radioactive Waste

The gaseous waste effluents from Units 2 and 3 will use a common 310-foot stack. Unit 1 will continue to discharge its gaseous effluents from its own 300-foot stack. The current stack release limits set forth in the Dresden Unit 1 Facility License DPR-2, as amended, will apply to aggregate releases from all three units. The radioactive effluent from each unit is monitored and filtered, and capability is provided for isolating the source from the environs. The summation of Unit 1, 2 and 3 stack gas radioactivity level will be indicated in each control room.

i. Inerting Gas Supply System

The inerting gas supply system for Units 2 and 3 is shared, although each unit has its own gas makeup system.

j. Common Turbine Building

The turbines for Units 2 and 3 are housed in a single turbine building. The turbine building ventilation air supply and exhaust systems are operated as a combined system.

k. Reactor Containment

Units 2 and 3 have separate and independent primary containments and pressure suppression systems. The secondary containment for each unit below the operating floor level is constructed to serve its own unit. The operating floor will be open and common to both units. Both units share the same standby gas treatment, ventilation and heating systems, each having capacities to accommodate the combined secondary containment volume.

l. Control Rooms

Evaluations have been made of the inter-plant effects of postulated accidents. It has been ascertained that an accident in any one of the units, up to and including maximum postulated accidents, will not prevent access to the control room of any of the units or prevent safe operation or shutdown of the other units.

Item	Reference Section in Plant Design and Analysis Report
High Pressure Coolant Injection System	Amendment No. 5
Low Pressure Coolant Injection System	Amendment No. 5
Emergency Coolant Supply	Amendment No. 5

7.0 EVALUATION OF PLANT SAFETY

The general safeguards objectives of the design of Unit 3 are to protect the plant equipment and to prevent radiation exposure in excess of a small fraction of established limits to any persons on or off the station premises, either during normal operation or during accident conditions.

In order to meet these objectives, the plant design and operation include the following:

a. Means for positive control of plant process parameters important to safety.

b. Inherent safety features and automatic devices are included in the design to prevent an operator error or equipment malfunction from causing an accident. Tests are conducted periodically to assure proper functioning of such devices.

c. Multiple barriers are provided to contain the radioactive materials. The core is conservatively designed to operated with thermal parameters significantly below those which could cause fuel damage.

d. The plant operating personnel are thoroughly knowledgeable in the plant operating characteristics, and are trained to follow written procedures to minimize the occurrence of operating errors.

This section summarizes the significance of these important features as they relate to safety of Dresden Unit 3.

7.1 Normal Plant Operation

Dresden Unit 3 is designed to operate safely under all normal operating modes for which the plant is designed. These operating modes include sustained full power operation, starting up, shutting down, load changing or maneuvering, refueling, hot standby, and shutdown. The plant is also designed to prevent radiation exposures in excess of a small fraction of acceptable levels to persons on or off the plant premises under accident conditions as well as the normal operating conditions.

This safety of operation is accomplished by incorporating specific engineered and inherent safety features in the design of the plant. The extent to which these features provide the desired levels of safety is summarized in the following paragraphs.

a. Control of Plant Power Output

During normal plant operations, the control of reactivity within the reactor core is accomplished by means of two basic power and reactivity control systems. Gross reactor power and power distributions within the core are controlled by the control rods which are manipulated by the operator. Reactor power can also be adjusted and controlled over a range of approximately 30 percent by variation of coolant recirculation flow rate through the core. These functions are accomplished by the operator from the control room.

The rate of power increase is limited by the rate at which control rods can be withdrawn, and the rate of variation of the recirculation flow. Control rods are operated one at a time and are withdrawn in a symmetrical pattern from the core. In this manner it is estimated that full load power operation can be achieved at a rate of approximately 3% per minute. This rate is sufficiently slow that load changes are always under control by the operator, and any maneuvering transients are accommodated well within the design parameters of the plant. Rapid shutdown is accomplished by insertion of the hydraulically actuated control rods.

The performance of the reactor core and the indication of reactor power level are continuously monitored by the neutron monitoring system, the sensors of which are located in the reactor core. This system efficiently and accurately provides power level monitoring from source range to full power on a gross and local basis.

The local power monitoring instrumentation is designed to monitor local fuel element heat flux continuously and provide information to permit evaluation of the critical core parameters.

The signals from the neutron monitoring system have the capability to initiate a scram, and this is accomplished through the logic circuit of the reactor protection system.

The reactor protection system also receives signals from other process parameters to shutdown the reactor when established limits are reached, and to initiate other protective functions as required.

The design of the reactor core nuclear performance characteristics include several features which contribute to a favorable nuclear dynamic response under transient conditions. The nuclear response characteristics provide strong negative reactivity feedback under severe transient conditions, contribute negative reactivity feedback consistent with the requirements of overall plant nuclear hydrodynamic stability, and provide a response which regulates or damps changes in power level and in the spatial distribution of power production in the core to a level consistent with safe and efficient operation. The response characteristics are inherent in the design of the plant and result from the negative doppler coefficient of reactivity of the fuel, the negative moderator temperature coefficient, and the negative

The above coefficients, together with the regulating or control devices, including the control rods, the flow control, and the initial pressure regulator, complement each other to achieve stable, well-controlled plant performance over the entire range of power operation.

b. Control of Core Cooling

The plant is provided with four methods of removing heat from the core. These heat removal systems, together with the core design features, provide assurance that adequate core cooling is available under all normal modes of plant operation, and under certain conditions of equipment malfunctions or operator errors, and include the following: (a) The shutdown cooling system is the normal mode of removing decay heat when the reactor is depressurized, (b) the isolation condensers with backup from the high pressure coolant injection syste, removes decay heat when the reactor is pressurized and isolated from other cooling systems, (c) the turbine and/or the main condenser is the principal heat sink when the reactor is at hot standby, is in full operation, or is starting up or shutting down, and (d) the relief vavles discharge steam to the suppression chamber under pressure transient conditions where it is condensed, and makeup water is added to the reactor by the feedwater system.

These systems are in addition to those which are used under certain accident conditions.

c. Control of Fuel Handling

The equipment and procedures used in the handling and storage of fuel are designed to permit safe, efficient refueling of the reactor. Replacement fuel is stored in a concrete vault in the reactor building. The vault is provided with radiation monitors, and the storage design prevents criticality even if the storage vault were flooded with water. All entrances to the vault, including fuel delivery doors and personnel openings, are capable of being locked.

The fuel pool for storage of spent fuel assemblies is also located in the reactor building. The racks in which the fuel assemblies are placed are bolted to the floor and are designed to ensure subcriticality in the pool. The pool water removes heat and shields against radioactivity, both of which are generated by the decay of fission products in the stored fuel. Transfer of irradiated fuel from the reactor core to the refueling pool is accomplished under water for control of airborne contamination. The new and spent fuel assemblies are handled by special hoists, cranes and grapples during the refueling or maintenance work. These include the building service crane, the refueling platform and hoists, reactor service platform, and the jib hoists. Each of these is designed with interlocks and special safety features to assure that fuel loading and handling can be conducted only under specific safe conditions.

d. Control of Radioactive Wastes

The radioactive waste control systems are designed to collect potentially radioactive wastes, and process and dispose of them in a safe manner without limiting plant operations or availability. Equipment, instrumentation, and operating procedures are designed so that the discharge of radioactive wastes will not exceed permissible levels. The liquid radioactive waste control system is designed to provide a means for processing potentially radioactive liquids, and to collect process liquid wastes in batches which may be sampled and analyzed to determine their suitability for release through the discharge canal into the river. Gaseous wastes from all three units are provided with a holdup time for decay then are filtered before being discharged to the atmosphere. It is anticipated that the total off-site dose from combined operation of all three plants will be less than one percent of the off-plant permissible annual dose.

e. Control of Radiation Levels and Personnel Exposure

Unit 3 is provided with two types of installed radiation monitoring systems which are utilized to maintain continuous surveillance and monitoring of the radiation levels associated with plant operations. The first type consists of several process radiation monitoring systems which provide a continuous indication and record of radioactivity at or near the discharge point of those process lines that can release radioactive effluents to the environs directly. These monitors are capable of measuring the radioactive material content of such effluents to a sufficient degree of accuracy to indicate that maximum permissible release rates are not being exceeded. Provisions are also made for monitoring the normally radioactive fluids which are contained or stored in the various plant process systems.

The second type is the area radiation monitoring system which provides a record of gamma radiation levels at selected locations within the various buildings, and the system is designed to alarm when radiation levels exceed preselected values. Radiation shielding is included in the plant design to minimize the exposure of plant personnel to radiation emanating from the reactor, turbine, and their auxiliary systems.

Additional shielding is provided so that the dose received by plant personnel in the control room in 8 hours, under design accident conditions, would not exceed 500 mrem.

7.2 Inherent Safety Features of the Reactor

The boiling water power reactor to be used in Unit 3 has inherent safety features derived from the materials, configuration, and operating modes specified in the design. Many plant equipment failures and/or plant maneuvers are accommodated by the reactor without violation of normal steady state operating limits. The reactor characteristics of importance in this regard are:

a. The characteristically large and virtually instantaneous negative Doppler coefficient of reactivity of slightly enriched uranium dioxide fuel provides an inherent mechanism for terminating nuclear transients.

b. The steam void coefficient of reactivity, as in the case of Doppler, also provides a negative reactivity feedback which tends to limit power excursions and contributes to the over-all plant stability. Voids and Doppler also tend to suppress peaks caused by improper rod withdrawal.

As an example of the benefits of the above nuclear characteristics, consider the loss of power to the pumps which supply the recirculation flow. If such loss were to occur at full power the reactor power would fall to a level consistent with the remaining natural circulation flow rate. No reactor scram would be necessary.

Another example of the self-limiting tendency of the reactor is the case of improper rod withdrawal during a startup.

In the event that improper rod withdrawal should produce a rising period as short as one second, the excursion would be terminated at a low power level by Doppler and void feedback and no scram would be necessary to protect the fuel. In this case, however, a scram does occur as the result of action by the intermediate range instrumentation.

The manner in which these inherent safety features and other design features contribute to normal safe operation of the plant were discussed above. Summarized below are the safety features incorporated in the several engineered safeguards and in the containment systems.

7.3 Engineered Safeguards

The normal plant control systems maintain plant variables within narrow operating limits. These systems are thoroughly engineered and backed up by a significant amount of experience in system design and plant operation. Even if an improbable operation error or equipment failure allows plant variables to exceed their operating limits, an extensive system of engineered safeguards terminates the transient and limits the effects to levels far below those which are of concern. Engineered safeguards include those which back up normal control systems, those which offer additional protection against a reactivity excursion, those which act to prevent a loss of coolant, and those which provide core cooling in the event of a loss of normal coolant. The containment and its cooling and inerting systems provide additional protection to the public.

A sufficient shutdown margin is maintained so that the reactor can be shut down in the cold condition with one control rod completely out of the core at any time in core life. The reactor can be shut down from operating power conditions with insertion of only a few rods because of the reduced reactivity due to hot fuel and moderator and to fission product neutron absorbers in the fuel. At any time during core life, many rods have to be withdrawn in order to bring the core critical.

In addition a completely redundant, separate and independent neutron absorbing shutdown system is supplied. The manually initiated standby liquid control system injects a neutron absorbing solution directly into the reactor moderator in quantity sufficient to shut the reactor down and maintain shutdown with the reactor in the thermally cold, fission product-free condition. This system would be used only in the abnormal case wherein control rods could not be inserted into the core.

The reactor protection system supplements the normal reactor power, pressure, water level, and radioactivity control systems. The reactor protection system automatically detects non-standard plant conditions and initiates reactor scram in time to provide protective margin against core damage. The system initiates reactor scram on high neutron flux, high pressure in either the reactor or drywell, high radiation in a steam line, or closure of the main steam line isolation valves among others. The equipment used in the reactor protection system, from the sensors through the circuitry and including the solenoid operated scram pilot valves in the control rod hydraulic system, is highly reliable and is duplicated for high probability of proper scram action when needed.

Abnormally high insertions of reactivity are prevented by the limitations on control rod drive withdrawal speed and control of the rate of moderator temperature change. The designs of the control rod to control rod drive coupling and the control rod drive thimble preclude an equipment failure which could allow the control rod to be removed from the core more rapidly than the control rod drive normal withdrawal mode. Nevertheless, three engineered safeguards have been included as supplementary protection. The movement of the control rod drive thimble and control rod is restricted by the thimble support, so that only negligible changes in core reactivity would result, even if the control rod drive thimble circumferentially ruptured. If a control rod were to become separated from its drive, stick within the core, and subsequently fall from the core, the "rod velocity limiter" would limit its free fall velocity. A rod worth minimizer computer limits the establishment of control rod configurations so that the maximum reactivity insertion rate resulting from the free fall of a control rod with a rod velocity limiter from the core would not cause damage to the primary system.

The piping of the reactor primary system is conservatively designed, analyzed, fabricated, and tested to prevent failure. Nevertheless, engineered safeguards have been provided to prevent release of harmful amounts of radioactive materials from the plant even if a pipe ruptures. If a pipe containing high pressure reactor water developed a significant rupture outside the primary containment, the rupture would be automatically detected and the isolation valves closed. The four large high pressure steam lines penetrating the containment have flow restrictors. If any such line were circumferentially ruptured, the flow restrictors would limit the coolant loss rate and the isolation valves would terminate the coolant loss so that no fuel damage would occur.

The reactor vessel and piping operate at 1000 psig, well below the design pressure of 1250 psig. The vessel and piping will be initially hydrostatically tested at 1560 psig. The initial pressure regulator, the bypass pressure regulator, the isolation condenser system, the relief valves, the safety valves and the reactor protection system operate to hold the system at operating pressure. Nevertheless, even if the piping ruptured inside the containment (where the coolant loss would not be terminated by closure of isolation valves), multiple independent coolant supply systems are provided to assure continuity of core cooling. These include either of two core spray systems, the high pressure coolant injection system, the low pressure coolant injection syste, and the emergency coolant supply system.

7.4 Containment

During all phases of normal plant operations, Unit 3 has no need for its containment systems. If the containment systems did not exist, the normal radiation exposures to the operating personnel and to the public would be no greater than those discussed above in Paragraph 7.1, "Normal Plant Operation." However, as a further backup safety feature, and in addition to the other engineered safeguards which are provided, the plant design includes a multiple barrier containment system whose primary function is to mitigate rapidly the consequences of postulated accidents involving the reactor and its various systems.

The primary containment design employs a pressure suppression containment system which houses the reactor vessel, the reactor coolant and recirculating loops, and other service loops connected to the reactor. The pressure suppression system consists of a drywell, a pressure suppression chamber which stores a large volume of water, a connecting vent system between the drywell and the water pool, isolation valves, containment cooling systems, and other service equipment.

The primary containment system is fabricated as two large pressure vessels and is designed to withstand the peak pressures which could occur due to the postulated rupture of any reactor primary system pipe inside the drywell. Containment spray cooling systems are provided to assure continuous cooling of the primary containment under accident conditions.

A reactor building completely encloses the reactor and its pressure suppression primary containment. This structure provides secondary containment when the primary containment is in service, and provides primary containment during periods when the pressure suppression containment system is open. From a safeguards consideration, the principal purpose of the secondary containment is to minimize ground level release of airborne radioactive materials and to provide for controlled, decontaminated elevated release of the building atmosphere under accident conditions.

8.0 SUMMARY OF OFF-SITE EFFECTS

Analyses have been made to evaluate the off-site effects from both normal plant operation and postulated accident conditions.

During normal full power operation of the plant it is anticipated that the maximum annual average exposure to persons off-site will not exceed approximately 5 mrems. This may be compared with the limitations of 10 CFR 20 of 500 mrems per year.

Two general classes of postulated accidents have been analyzed; namely, reactivity transient accidents typified by the rod drop and fuel loading accidents, and component failure accidents such as the steam line break and a loss of coolant. The following summarizes the maximum calculated personnel doses at the site boundary for the above accidents.

Accident	Total Accident Exposure - Rem	
	Whole Body	Thyroid
Rod Drop	3.1×10^{-2}	1.7×10^{-2}
Fuel Loading	3.2×10^{-2}	2.8×10^{-1}
Steam Line Break	6.8×10^{-6}	3.2×10^{-2}
Loss of Coolant	4.2×10^{-4}	5.4×10^{-4}

It has been concluded from these analyses that the off-site exposure levels are only a small fraction of the maximum recommended guideline values of 10 CFR 100, and thus the plant provides substantial and adequate protection against hazards to the public.

9.0 Conclusion

On the basis of the foregoing the Applicant respectfully submits that:

a. A description of the proposed esign of Unit 3, including the principal architectural and engineering criteria therefor, has been furnished;

b. Applicant has identified the major new features incorporated in the design of Unit 3 which have not been demonstrated in actual operation of other nuclear power plants;

c. There are no unresolved safety questions other than those related to the demonstration of the effectiveness and reliability of such new features by actual operation;

d. The effectiveness and reliability of such new features will be demonstrated by the operation of other similar boiling water reactor plants now under construction and by pre-operational and initial startup tests of Unit 3 prior to the last date stated in the application for completion of construction of such unit; and

e. Taking into consideration the characteristics of the site and environs and the proposed design of Unit 3, such facility can be constructed and operated within the limitations established by 10 CFR 20, within the site criteria set forth in 10 CFR 100 and without undue risk to the health and safety of the public.

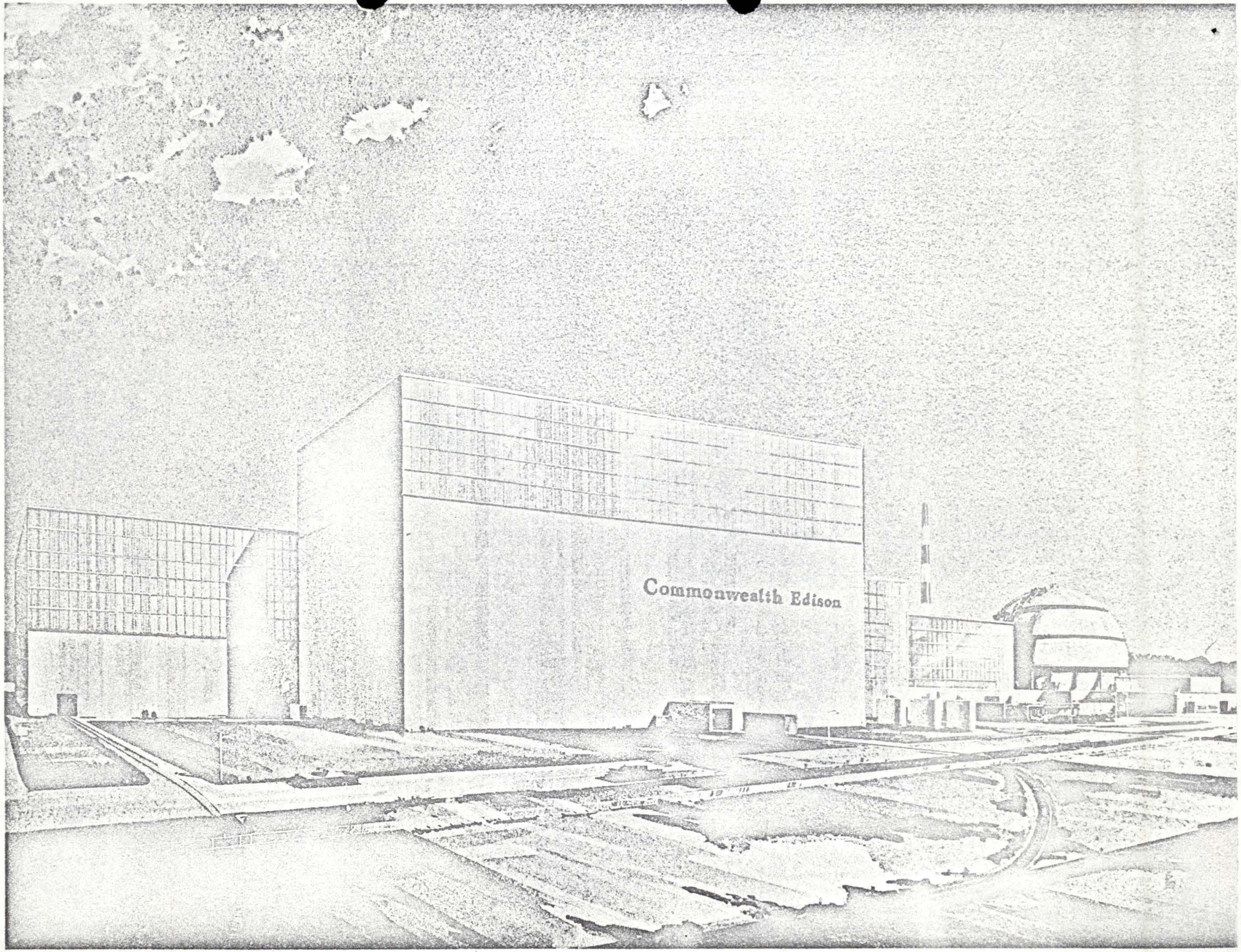

Plate 1. Station Perspective

INDEX TO SECTIONS OF
UNIT 3 PLANT DESIGN AND ANALYSIS REPORT
IN WHICH UNIT 3 DESIGN DESCRIPTIONS AND
ANALYSES DIFFERS FROM THOSE REVIEWED IN
AEC DOCKET 50-237

A. Changes in Interconnections and Sharing of Auxiliary Systems and Common Facilities

Item	Section Reference in Unit 3 Plant Design and Analysis Report
General Description	Section I-7
Reactor Building (Secondary Containment)	Section V-4
Turbine Building	Section I-4
Radwaste Building	Section VI-2
New Fuel Storage	Section IX-1
Spent Fuel Shipping Facility	Section IX-1
Service Water System	Section IX-5
Reactor Building Cooling Water System	Section IX-5
Turbine Building Cooling Water System	Section IX-5
Reactor Building Heating & Ventilation System	Section V-4
Turbine Building Heating & Ventilation System	Section I-7
Standby Gas Treatment System	Section V-4
Make-up Water System	Section IX-5
Make-up Demineralizer	Section IX-5
Fire Protection System	Section IX-5
Circulating Water System	Section IX-2
Radioactive Waste Control Systems	
Liquid Waste Control	Section VI-2
Gaseous Waste Control	Section VI-3
Solid Waste Control	Section VI-4
Standby Power Sources	
34.5 KV Line	Section VII-1
2500 KVA Standby Transformer	Section VII-2
Standby Diesel Generator System	Amendment No. 5
Control Room	Section IX-4

B. Changes Related to the Provision of a Turbine Generator Approximately 2% Larger in Capacity Than Originally Considered

Item	Section Reference in Unit 3 Plant Design and Analysis Report
Turbine	Section IX-2

Item	Section Reference in Unit 3 Plant Design and Analysis Report
Reactor Vessel Overall Length Inside	Section I-4
Fuel Rod Pitch	Section IV-1
Reactor Recirculating Loops Design Pressures	Section V-2.3
Primary System Hydrostatic Test	Section V-2.6
Primary Containment System	Section V-3
Drywell External Design Pressure	
Downcomer Vent Pressure Loss Factor	
Drywell Free Volume	
Pressure Suppression Chamber Free Volume	
Recirculation Line Pipe Supports	Section V-3.2
Drywell NDT	Section V-3.2
Pressure Suppression Chamber NDT	Section V-3.3
Penetrations of Primary Containment and Isolation Valves	Section V-3.4 and V-3.5
Containment Inerting System	Section V-3.7
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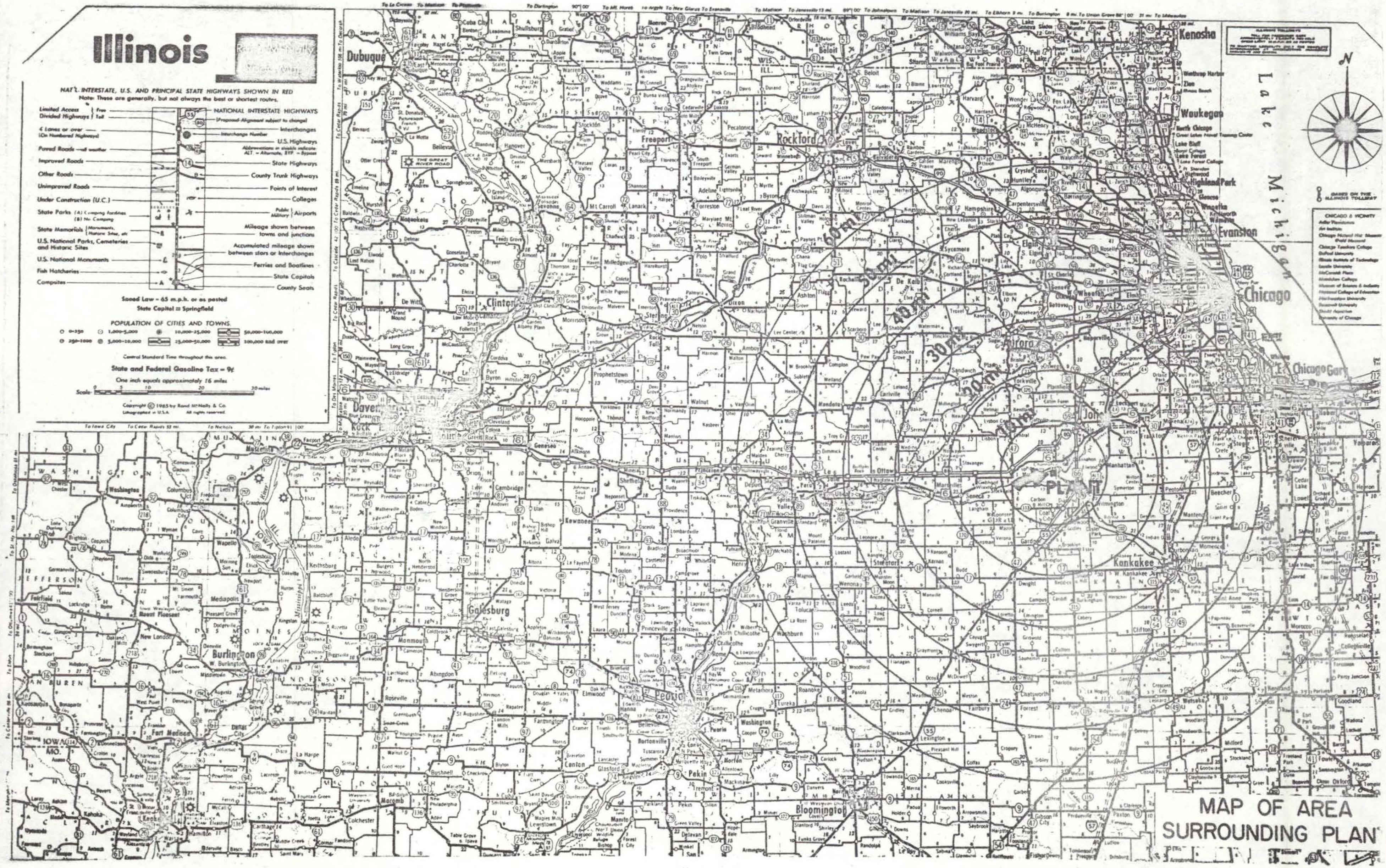
Isolation Condenser System	Section V-5.2.3 and Amendment 5
High Pressure Coolant Injection System	Amendment 5
Core Spray Systems	Section J-3.7.3b and Amendment 5
Low Pressure Coolant Injection System	Amendment 5
Containment Spray Cooling Systems	Amendment 5
Emergency Coolant Supply	Amendment 5
Automatic Relief Valve Actuation	Amendment 5

D. Refinements in Certain Analyses

Item	Section Reference in Unit 3 Plant Design and Analysis Report
Inadvertent Isolation Valve Closure	Section XI-2.5.1
Loss of Auxiliary Power	Section XI-2.7.3
TIP System Guide Tube Failure	Section XI-2.7.7
Control Rod Drop - Radiological Effects	Section XI-3.1.6
Fuel Loading Accident	Section XI-3.2
Steam Line Rupture Outside Reactor Building	Section XI-3.3
Loss-of-Coolant Inside Drywell	Section V-3.7.2 Section XI-3.4 Section XI-3.5.4
Analytical Methods	Section XI-4


Illinois
NAT'L. INTERSTATE, U.S. AND PRINCIPAL STATE HIGHWAYS SHOWN IN RED
Note: These are generally, but not always the best or shortest routes.
POPULATION OF CITIES AND TOWNS
State and Federal Gasoline Tax = 9¢
One inch equals approximately 16 miles
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Joliet
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Waukegan
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MAP OF AREA
SURROUNDING PLAN'

PLATE 3

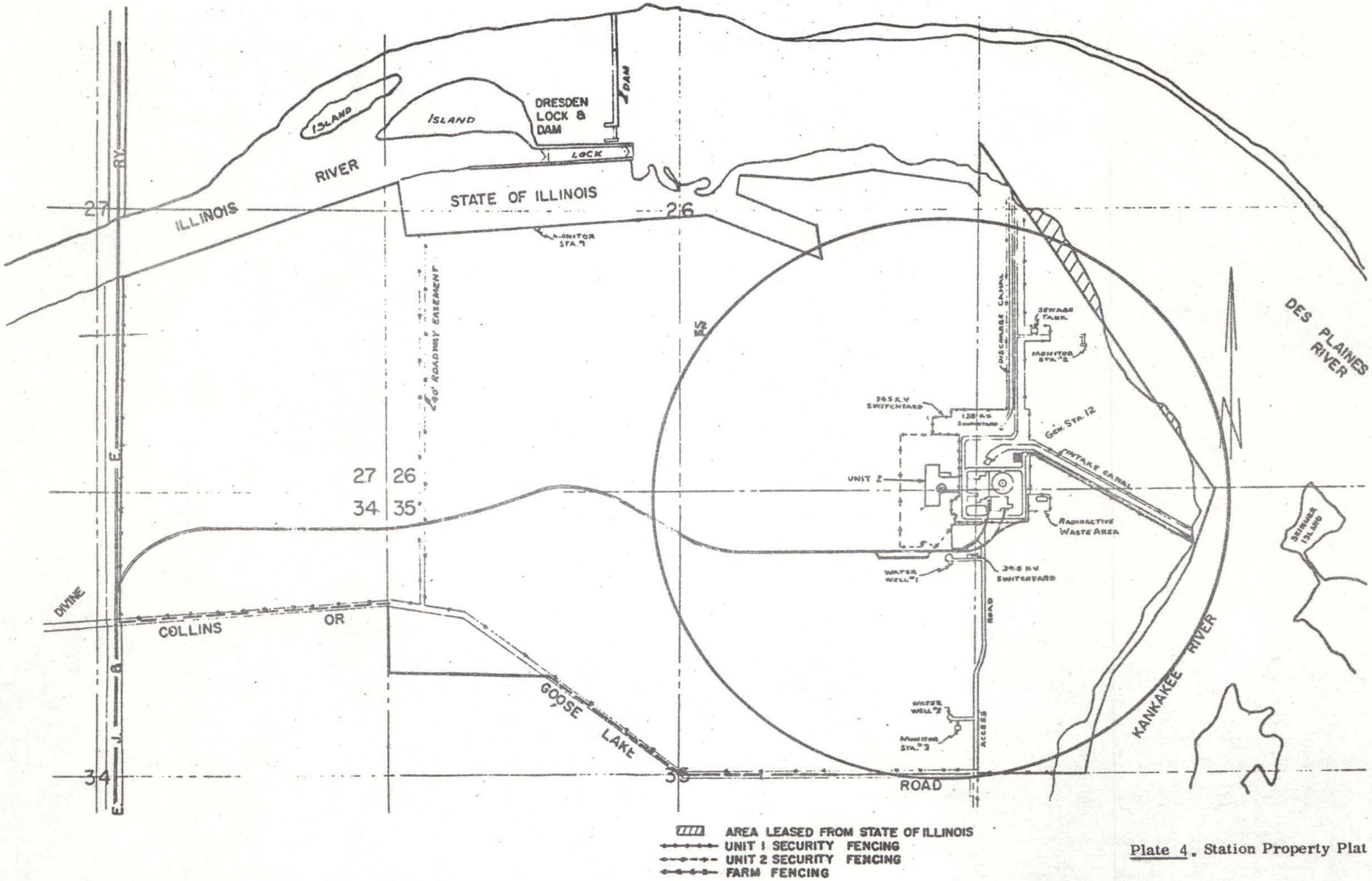

Plate 4. Station Property Plat

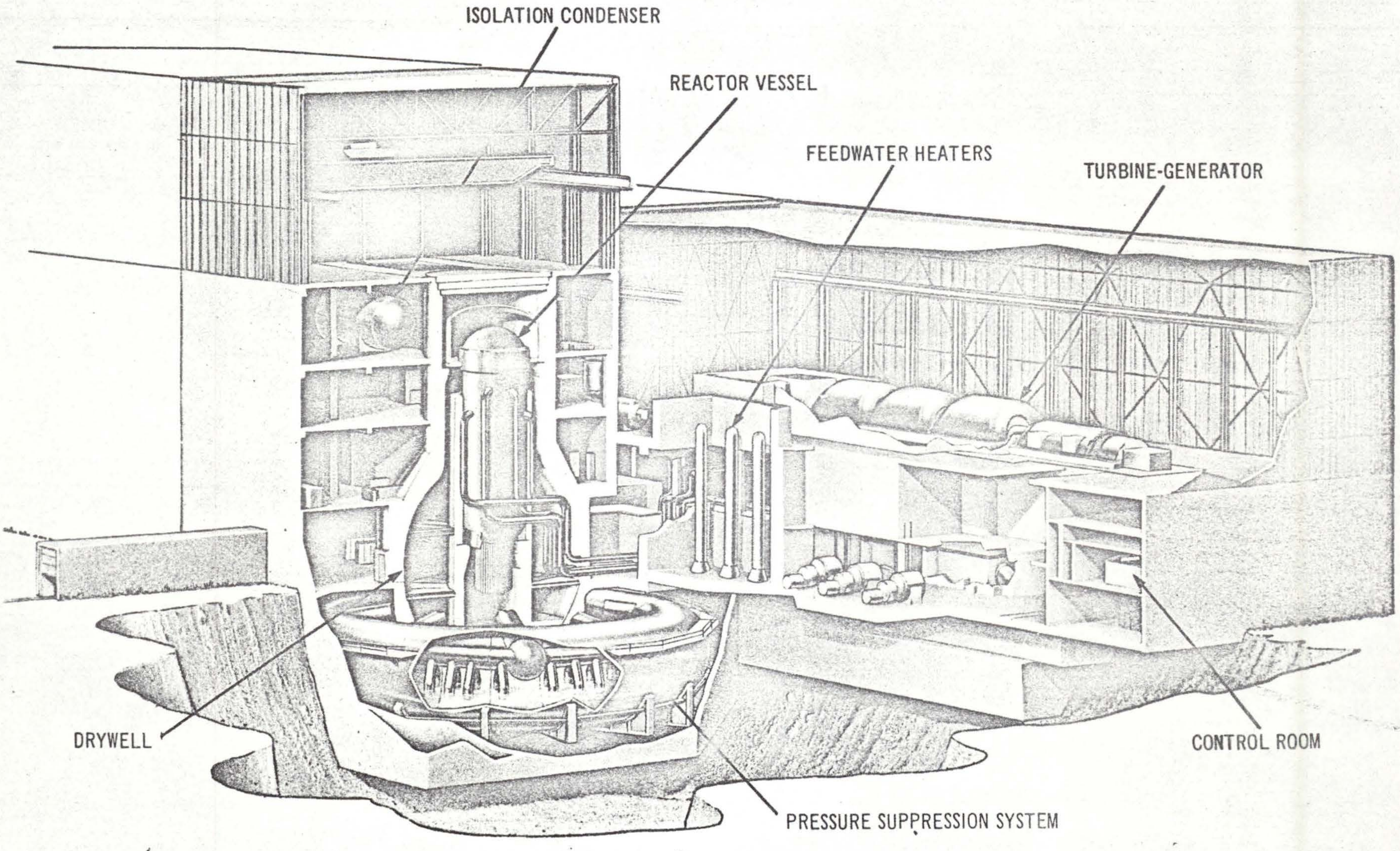

Plate 5. Artist's Conception of Typical Reactor and Containment

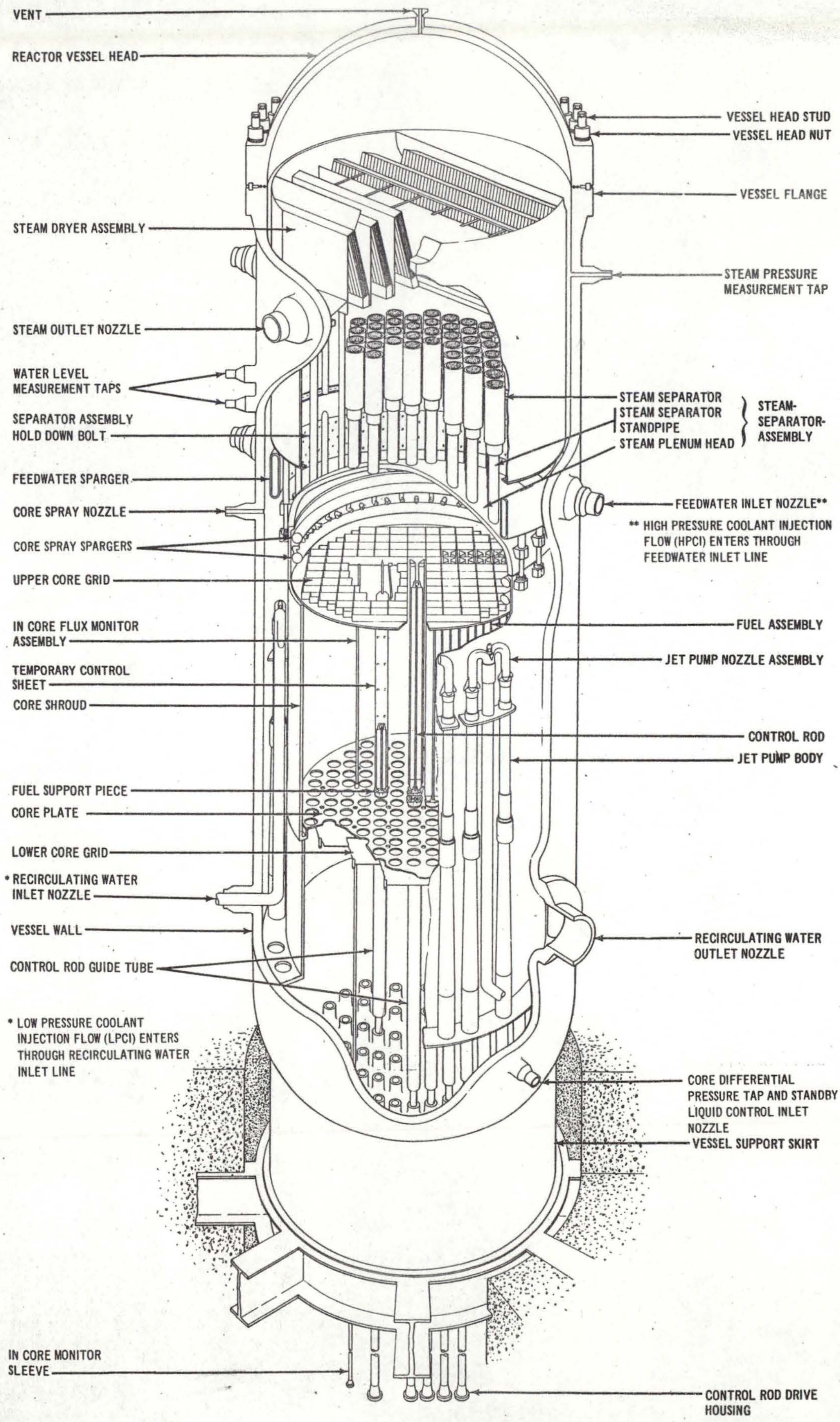

Plate 6. Reactor Vessel and Core Arrangement