November 24, 1965

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

IN THE MATTER OF

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION UNIT 2

GRUNDY COUNTY, ILLINOIS

DOCKET NO. 50-237

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I. Introduction

On April 15, 1965, the Commonwealth Edison Company applied for a license to construct and operate a 2255 megawatt thermal (Mwt) nuclear facility to be located at the Dresden Nuclear Power Station in Grundy County, Illinois in accordance with the Atomic Energy Act and the Commission's regulations. The issues to be considered, and on which an affirmative finding must be made in order to issue the license requested, are set forth in the Notice of Hearing issued by the Commission and published in the Federal Register on October 27, 1965.

The technical review of the proposed design of the Unit, which has been conducted by the staff of the Commission's Division of Reactor Licensing, is based on the report, Dresden Nuclear Power Station Unit 2 Plant Design and Analysis Report, and five amendments thereto. The staff has also held a number of meetings with representatives of the applicant and General Electric to discuss and clarify the material submitted. The Commission's Advisory Committee on Reactor Safeguards (ACRS) also considered this project and has reported its views to the Commission by letter, a copy of which is attached as Appendix A.

Upon completion of construction, an operating license for the proposed facility would be issued only after a thorough evaluation of the final design of the facility by the Commission, and after the Commission finds that all its safety requirements have been met. Finally, when authorized to operate, the plant would be operated in accordance with the Commission's regulations under the scrutiny of the staff. The construction permit is thus the first step of a regulatory process which will continue throughout the lifetime of the facility.

II. Site Description

The site proposed for Dresden Nuclear Power Station Unit 2 comprises approximately 953 acres owned by Commonwealth Edison Company and located in Grundy County, Illinois at the junction of the Kankakee River with the Des Plaines River. This site is approximately 8 miles east of Morris and 14 miles southwest of Joliet. The nearest boundary of the site is approximately one-half mile to the south of the proposed location of Unit 2. Thus, the minimum exclusion distance is approximately 1/2 mile.

A. Population Density

The average population density within five miles of the site is 32 per square mile. Within five to ten miles there is a population of 20,000 averaging 85 persons per square mile. The largest town within ten miles is Morris (population 7,935). Within 10 to 15 miles the population is estimated to be 80,000 with the bulk contributed by Joliet (population 67,000). Based on this information, the outer boundary of the low population zone, as defined in 10 CFR 100, is approximately 10 miles from the site. The nearest population center with a population of more than 25,000 is Joliet, about 14 miles from the site.

B. Meteorology

The applicant furnished a description of the meteorological environment for the proposed site prepared by a meteorological consulting firm based on climatological records from U. S. Weather Bureau stations in the area and on data collected at the site and at Argonne National Laboratory.

Although the diffusion climatology of this site has not as yet been established in detail, the general climatology of the area can clearly be conservatively represented by use of those parameters suggested in TID-14844 (Calculation of Distance Factors for Power and Test Reactor Sites). These parameters have been used for estimating the potential consequences of accidents in subsequent sections of this report. The present gaseous effluent release limit via the facility stack of Dresden Unit 1, which will be used in common by both Units 1 and 2, is 0.7 curies/second. This limit, or that which will be specified at the time of granting the operating license for this facility, will be the total allowable release from both units. In either case, the off-site doses resulting from this source will be within those specified in 10 CFR 20, the Commission's standard for protection against radiation. The diffusion climatology is adequate to support this limit. The U. S. Weather Bureau has reviewed the meteorological information included in the report. Its comments are attached as Appendix B.

Of significance with respect to safety design considerations are the maximum wind speeds associated with the severe weather conditions. The maximum wind velocity reported in the area of the site is 109 miles per hour, unofficially reported at Joliet. The applicant is proposing a design wind speed of 110 miles per hour for all structures important to safety, which compares favorably with the maximum wind speed reported for the area. The site is also susceptible to tornado activity. The maximum wind velocity during a tornado could be expected to exceed the design value selected. However, the applicant has considered the possibility for a tornado striking the plant and has proposed design features adequate for protection of those components vital to reactor safety should such an event occur.

C. Geology and Hydrology

The information described by the applicant indicates that geologic and hydrologic conditions of the Dresden site are not expected to present any unusual problems with respect to design and construction of the proposed facility. A report has been prepared by the U. S. Geological Survey which supports this conclusion. A copy of the report is attached as Appendix C. The Fish and Wildlife Service has also reported on the related aspects of Unit 2 and concluded that plans for control and disposal of radioactive liquid waste are adequate to protect fish and wildlife in the vicinity of the proposed Unit. This report is attached as Appendix D.

D. Seismology and Seismic Design

The Dresden site is located in Zone 1 (zone of minor damage) on the seismic probability map of the 1958 Uniform Building Code. In this regard the applicant proposes that structures and equipment important to safety will be designed in conformance with the following criteria:

(1) Functional loading in combination with an acceleration spectrum corresponding to a maximum ground acceleration of 0.1 g shall be within allowable working stresses, and

(2) Combined stresses from functional and seismic loading of these structures will be such that a safe shutdown would be assured if subjected to a maximum ground acceleration of 0.2 g.

A report on the seismicity of the Dresden site provided by the U. S. Coast and Geodetic Survey confirms that the 0.2 g ground acceleration is more than adequate for the Dresden site. The U. S. Coast and Geodet

Survey report is attached as Appendix E. The favorable report of our seismic design consultant, Nathan M. Nemark, is attached as Appendix F.

III. Facility Design

It is proposed that Dresden Unit 2 will be a 2255 thermal megawatt (Mwt) boiling water, single cycle unit from which 715 electrical megawatts (Mwe) will be generated. The nuclear reactor is fueled with uranium dioxide (UO_2) sealed in Zircaloy rods. Individual fuel rods are assembled into fuel elements of which 724 are assembled into the active reactor core. The active core is about 12 feet long and about 15 feet in diameter. The core is contained in a pressure vessel which is about 68 feet long by 21 feet in diameter. Cooling water is circulated through the core by two 45,000 gpm recirculation pumps plus 20 jet pumps. The steam formed in the core is dried by in-vessel moisture separators and then piped to the turbine-generator.

The reactor vessel and recirculating pumps and motors are located within a pressure suppression containment structure similar to that installed at the Humboldt Bay facility and under construction for Jersey Central and Niagara Mohawk. This containment structure is designed for low leakage to retain fission products which might be released as a result of an accident. The containment structure is located within a reactor building. This building is of a controlled leakage design such that during an accident situation, leakage from the containment structure would be directed to the facility stack via a filtering system.

Emergency cooling systems are installed within the reactor pressure vessel and within the containment structure to supply cooling as required to protect the nuclear core and the containment structure from serious damage in the unlikely event of a major loss of coolant accident.

Inasmuch as the applicant has provided extensive details concerning the design of the facility in the Plant Design and Analysis Report, the staff does not believe that any useful purpose would be served by further repetition in this analysis.

IV. <u>Comparison of Dresden Unit 2 with The Jersey Central and The Niagara Mohawk Facilities</u>

The proposed facility is of the same design as the Jersey Central and Niagara Mohawk facilities, except for the following significant differences:

1. Jersey Central and Niagara Mohawk facilities are to be operated at power levels of about 1600 Mw(t). The proposed Dresden Unit 2 is rated at 2255 Mw(t). This approximate 50% increase in power level is accomplished by increasing the core size from 500 to 724 fuel assemblies. The therm characteristics of the core are not substantially changed.

The principal question to be considered is whether there is some safety significance connected with the increase in core size. Our review indicates that the most likely problem would be some form of instability. A summary of the analytical work involved in calculating the dynamic response of the reactor nuclear, thermal, hydraulic system is given in answer to Question II-2 in Amendment No. 2 to the application. We believe that the extent of analysis to be conducted which has been outlined, and the stability criteria stated in this answer is sufficient to provide assurance that no safety problems are likely. By the time Dresden Unit 2 is to operate, there will be operating experience with the KRB boiling reactor in Germany (800 Mw(t)) and with Jersey Central and Niagara Mohawk (about 1600 Mw(t)). The dynamic characteristics of these reactors will be measured. The interpretation of these data should aid in interpretation of the dynamic characteristics of the Dresden Unit 2 reactor.

2. Recirculation flow in Dresden Unit 2 will be by two 45,000 gpm capacity pumps. Each will be connected to 10 jet pumps arranged in parallel and situated in an annulus between the reactor vessel wall and the thermal shield. Based on the information presented by Commonwealth on the characteristics of the jet pumps as applied to recirculation flow control, it appears that the recirculation flow control properties will differ little from those of Jersey Central and Niagara Mohawk. Reactor power can be varied over an approximately 30% range by varying recirculation flow between 70% and 100%. As with Jersey Central and Niagara Mohawk an interlock to prevent control rod withdrawal outside an acceptable power-flow range will be provided. This interlock is designed to prevent fuel damage from the combination of low recirculation flow and high power.

Recirculation flow, and thus reactor power, will be adjusted either manually by a signal from the operator or automatically by a load error signal.

A significant amount of testing of jet pumps in single as well as multiple units has been performed to date to develop an optimum design. It is planned to instrument the pumps and associated equipment so that further testing and diagnostic measurements can be performed when installed in the reactor. Such measurements are necessary since, although the characteristics of single full-size jet pumps have been measured, the characteristics of full-size jet pumps in multiple set-ups have not been measured.

The final assessment of the effects of jet pumps on plant stability will be made during preoperational testing of the Unit. Instability between the five 14-inch lines entering the reactor downstream of a recirculation pump would be unlikely because of the relatively high resistance of these lines and the jet pump nozzles. Some instability between the individual pairs of jet pumps may be a problem which the applicant intends to investigate during preoperational testing. Tests of scale models operated in groups, however, show no tendency toward instability even when efforts were made to drive the pumps to instability.

Based on the foregoing discussion, and on the material presented describing the characteristics of jet pumps and the planned preoperatio testing program, we believe that jet pumps as proposed can be developed and used safely.

3. Dresden Unit 2 will be on the same site as Dresden Unit 1. The two units will be completely independent except for common use of the 300-foot tall facility stack for effluent releases, the intake and discharge canals for cooling water and effluent releases, the administrative and service facilities, and the 138 KV reserve auxiliary power. None of these are of such a nature that failure would cause an accident simultaneously involving Unit 1 and Unit 2, nor are they of such a nature that an accident at one unit would propagate to the other.

The stack release limit, currently 0.7 curies/second for Unit 1, will apply to the aggregate release from Units 1 and 2. The liquid effluent release from the discharge canal will remain in accordance with 10 CFR

We believe that the two units will be sufficiently independent so that interaction of the two units need not be a safety consideration. The exception to this is control room shielding. This is considered in a subsequent section of this report.

4. It is proposed that all nuclear instrumentation to be used with Unit 2 be installed in-core.

The startup instrumentation consists of four retractable, miniature fission chambers which generate Log Count Rate and period information. These channels are not connected to the reactor protection system and thus will not provide a scram function. The LCR signals actuate several interlocks relating to fission chamber and control rod withdrawal. Period signals are fed to annunciators. With retraction, the startup range instrumentation covers the flux range from source level to approximately 10% rated power.

The intermediate range instrumentation consists of eight "Campbell" channels, each of which ultimately feeds a variable range picoammeter. Each channel has a range from 40 x 10^{-5}% to 125% rated power. Scram signals are generated and fed to the dual bus scram channels of the reactor protection system whenever the indicated reading at a picoammeter exceeds a certain percentage of full scale reading, regardless of actual power being measured. Various rod and chamber withdrawal interlock functions are also provided.

The power range instrumentation consists of one hundred sixty-four miniature fission chambers, each of which has its own individual readout meter. Collectively, these channels are known as the Local Power Range Monitoring System (LPRM). Sixty-four of these channels are also connected to what is known as the Average Power Range Monitoring System (APRM). There are eight APRM amplifier units to each of which, respectively, are connected eight of the sixty-four channels. The output of each APRM amplifier unit is a signal proportional to the average of all signals from eight channels. These signals trip the dual-bus scram channels of the reactor protection system when level set points are exceeded.

Each intermediate range channel and each APRM unit individually induces a rod-block action in response to high flux.

General Electric is conducting a development program of in-core instrumentation including in-core testing of prototype chambers in Dresden Unit 1.

A traversing-in-core-probe (TIP) system is used to calibrate the fixed (power range) detectors. Each of five subsystems consists of a chamber, identical to those used in the power range, with an integral, flexible motor driven cable. Motors located in the reactor building drive and cables mechanically, with the detectors at the tips, through guide tubes which penetrate the containment, into the reactor core. This differs from the instrumentation initially proposed for Jersey Central and Niagara Mohawk; however, we understand that both of these facilities will also use in-core instrumentation.

We have reviewed the proposed in-core instrumentation by considering its ability to provide the required information at least as reliably as out-of-core instrumentation. Based on our review and the development program proposed, we believe that a suitable in-core instrumentation system can be designed to perform as required.

V. Conformance of Dresden Unit 2 Design to Staff's General Criteria

The following detailed safety analysis of Dresden Unit 2 has been organized in the framework of criteria which have been developing over the years and have been used by the staff in its evaluation of applications for power reactors. This is done for reasons of clarity and convenience as well as to demonstrate the applicability of the criteria to the safety evaluation of the proposed nuclear power plant.

FACILITY

CRITERION 1

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. It should be recognized, in this respect, that design codes commonly used for non-nuclear applications may not be adequate.

(b) Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice, and other natural phenomena anticipated at the proposed site.

The applicant states that the reactor primary system components will be designed and fabricated in accordance with the applicable ASME Boiler and Pressure Vessel Code, Section III Nuclear Vessels, or the ASA Code for Pressure Piping. Although the operating pressure of the primary system is to be 1,000 psig, the design pressure is 1250 psig. This differential is to accommodate maneuvering transients without safety valve operation. The pressure vessel will be the largest yet used in a nuclear application. We believe that particular care should be taken during the detailed design, fabrication and operation of the vessel to insure that any potential problems due to its large size are identified and taken into consideration.

The design temperature for the various primary system components will depend upon the specific expected temperature plus that due to radiation heating.

The drywell, the suppression chamber vent pipes, and the suppression chamber will be designed and fabricated in accordance with the appropriate sections of the ASME Pressure Vessel Code, Section III. Each will be designed for a pressure of 62 psig at an internal temperature of 281°F.

The engineered safeguards, consisting of the drywell cooling equipment, the core spray systems, and the containment cooling systems will all be designed in accordance with the applicable ASA Code for Pressure Piping.

The instrumentation, control system, and safety interlock components will be "reactor grade," that is, fabricated of the highest quality materials and workmanship. The same high standards of quality which are applied to the safety systems will also be applied to the safety interlock equipment.

The secondary containment system, the reactor building, will be designed to withstand the loading associated with a sustained wind velocity of 110 mph. This, we believe, is adequate except for the case of tornadoes which are discussed below. Flooding can be excluded as a consideration since the principal structures will be located at least 10 ft. above the maximum historical flood elevation of 506 ft. msl.

Seismic and tornado design of the facility have been given detailed consideration. As has been previous practice, two seismic design accelerations, 0.1 g and 0.2 g, have been specified. The facility will be designed so that the material of Class I structures (structures and equipment important to safety) when subjected to an acceleration spectrum corresponding to 0.1 g will not exceed yield stress under combined functional and seismic loading stresses.

In addition, the facility will be designed so that function of Class I structures will be assured following a 0.2 g earthquake. This includes, for the case of the containment, combined accident and seismic loads. The U.S.C. & G.S. has reviewed these accelerations and concluded: ". . . ground accelerations of more than 0.1 g . . . will not be encountered at the Dresden site. . .". Our seismic design consultant has reviewed the material presented in the application concerning seismic design and has concluded "On the basis of the information with which we have been supplied, we believe the design criteria outlined for the primary containment, secondary containment structures, and Type I piping, will provide an adequate margin of safety for seismic resistance."

The basis for the tornado design of the facility is that the likelihood of reinforced concrete structures being damaged is low. Thus the facility is designed with all components required for safe shutdown either within a reinforced concrete structure or below ground. The electrical transmission system might be a vulnerable component; however, with the number of transmission lines (7 entering from two directions), and the fact that one of the lines is underground near the facility we believe that simultaneous loss is unlikely. Even so, an emergency diesel generator which Commonwealth states will be started when there is a tornado alert, is available and will be situated in a reinforced concrete block structure.

Based on these considerations, we believe that this criterion is satisfied.

CRITERION 2

Provisions must be included to limit the extent and the consequences of credible chemical reactions that could cause or materially augment the release of significant amounts of fission products from the facility.

Assuming a loss of coolant accident in the case of Dresden Unit 2, the most severe credible chemical reaction would be a zirconium-water reaction since the fuel cladding and the fuel assembly channel pieces will be fabricated of Zircaloy-2.

The applicant calculates that if the reaction terminates when the zirconium reaches 3300°F, its melting point, a 24.5% Zr-H_2O reaction would occur assuming (1) just enough water available to continue the reaction and (2) the system is adiabatic. Termination at 3300°F is presently assumed because at this temperature it is expected that the cladding would slump, block channels, and impede the entrance of steam. However, core meltdown would continue unless some cooling were available, and as long as meltdown is continuing, it is not possible to predict the ultimate course of subsequent metal-water reactions. General Electric has already conducted an extensive program to determine the course of potential metal-water reactions, and this program will be continued. The significance of this matter is discussed in the context of the consequence of a loss of coolant accident in the accident analysis section of this report.

Provisions which would limit the extent and consequences of such reactions include the core spray system. Each of the independent spray loops will be capable of supplying water to the core region and would re-cover the core in about 3 minutes. The system is actuated by low primary pressure and low reactor water level signals which under present design requirements will begin to refill the core region after the primary system pressure had dropped to 150 psig. The present design flow capacity of each loop is 4400 gpm and back-up pumping capacity will be available. The General Electric Company intends to give further study to the time sequence of events which would result from a major loss of coolant accident in order to assure that the core spray system to be installed will provide an adequate and timely emergency cooling capability to prevent any significant fuel melting under these conditions. The core reflooding capability, effectively a watertight vessel within the pressure vessel, will allow the core to be covered up to 2/3 to 3/4 of the active fuel height. After this, makeup due to boil-off is all that would be required to maintain adequate cooling. The applicant has calculated, and we agree, that if the equipment functions as designed, less than 1% Zr-H_2O reaction would occur from a major loss of coolant accident. This amount of reaction is negligible.

Assuming, however, that the core spray system does not function, a Zr-H_2O reaction as low as 4%, along with recombination of the hydrogen and oxygen, could jeopardize the containment integrity. Because of this, the applicant has stated that the containment will be inerted and the oxygen content limited to less than 5% by volume. Inerting will prevent recombination of hydrogen and oxygen.

We believe that the systems and precautions discussed above satisfy this criterion. (See also the discussion under Criterion 17).

CRITERION 3

Protection must be provided against possibilities for damage of the safeguarding features of the facility by missiles generated through equipment failures inside the containment.

A drywell wall design criterion is: "To withstand a jet force equal to that associated with flow from the largest local pipe or connection without containment failure." Jet forces are calculated by assuming that reactor pressure acts directly on the containment over an area equal to that of the largest diameter local pipe or nozzle. With reference to missiles, the applicant intends to maintain a constant surveillance on equipment design so that if a missile appears to be credible, appropriate action will be taken to protect the containment. For all except the primary coolant system, we believe that this approach is reasonable and that the criterion is satisfied.

Considering the primary system, the applicant states ". . . it has been concluded that with the application of conservative piping design and process engineering practices, pipes will not break in such a manner as to bring about movement of pipes sufficient to damage the primary containment vessel." Notwithstanding the applicant's conclusion, we believe that the primary system pipes and supports should be designed to withstand, without excessive motion which could jeopardize containment integrity, whatever reaction forces would be associated with the applicant's design basis accident for the containment, namely a complete severence of a recirculation line. In this respect, at the present time, Criterion 3 may not be satisfied; however, the design of appropriate pipe supports is within the realm of standard engineering practices, and therefore, Criterion 3 can be satisfied during the detailed design of the facility.

CRITERION 4

The reactor must be designed to accommodate, without fuel failure or primary system damage, deviations from steady state norm that might be occasioned by abnormal yet anticipated transient events such as tripping of the turbine-generator and loss of power to the reactor recirculation system pumps.

Primary protection against possible fuel failure or primary system damage is provided by the reactor protection system and the multiple heat sinks. The operating parameters of the reactor system are continuously monitored, and a scram is initiated before any safety limit is exceeded. The scrammed control rods will shut down the reactor, and the stored and decay heat can be dissipated by either the turbine condenser or the isolation condensers. The primary heat sink for the reactor is the turbine condenser. If this becomes unavailable due to any combination of steam valve failures (isolation, throttle, stop, and bypass valves), the reactor heat can be dissipated in the two isolation condensers.

Inadvertent steam line isolation valve closure would result in a reactor scram, loss of the turbine condenser as a heat sink, and subsequent increase of the reactor pressure to the isolation condenser trip point of 60 psi overpressure within about 40 seconds. At this point the cooling provided by the isolation condenser would terminate the pressure transient.

Turbine trip with proper function of the 40% rated steam flow capacity bypass valves would result in a reactor scram, a neutron flux spike to 140% rated, and a pressure transient peaking at about 70 psi above operating. Depending on the severity of the pressure transient the operation of the isolation condensers may or may not be required.

A turbine trip accompanied by failure of the bypass valves to open would result in a reactor scram, a neutron flux spike to 200% rated, and a pressure transient peaking at 120 psi above operating pressure. The relief valves which function at 100 psi above operating would function but the safety valves would not. The fuel rod surface heat flux transient for the turbine trip incidents would be negigible due to the relatively long time constant of the fuel.

Loss of recirculation pump power would result in a reactor scram, a gradual coast-down of the circulation flow, and heat flux spike resulting in a maximum critical heat flux ratio (MCHFR) of 2.0. Shaft seizure and consequent immediate stoppage of flow in one recirculation loop would cause a heat flux spike resulting in a MCHFR of 1.3. We believe that this is acceptable for such improbable equipment malfunctions.

Based on the foregoing, we believe this criterion is satisfied.

CRITERION 5

The reactor must be designed so that power or process variable oscillations or transients that could cause fuel failure or primary system damage are not possible or can be readily suppressed.

The applicant has performed a large number of analytical studies to determine the stability characteristics of the reactor nuclear, thermal, and hydraulic systems. These studies analyzed potential transients induced by changes in pressure, recirculation flow, subcooling, and control rod position. This work has indicated that divergent power oscillations induced by process variations are not anticipated.

The scram system has been designed to shut down the reactor safely if a turbine trip were to occur from rated power. The resulting transient would cause the neutron flux to increase by approximately 100% per second. By comparison, the most severe oscillatory transient which the applicant believes could occur would result in a maximum neutron flux rise of about 8% per second. However, since no power reactor of this size has been constructed or operated, the possibility of power oscillations induced by some unforeseen mechanism cannot be completely precluded. The preoperational and startup testing programs, however, should enable any potential unstable conditions to be recognized and corrected.

We believe that the design and analysis of the reactor system meets the criterion.

CRITERION 6

Clad fuel must be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without experiencing significant cladding failures. Unclad or vented fuels must be designed with the similar objective of providing control over fission products. For unclad and vented solid fuels, normal and abnormal modes of anticipated reactor operation must be achieved without exceeding design release rates of fission products from the fuel over core lifetime.

The reactor fuel elements will be 12 feet in length and contain UO_2 fuel having an average enrichment of 2.0 weight percent U-235. The fuel rods will be clad with Zircaloy having an outside diameter of 0.570 inch and wall thickness of 0.036 inch. The minimum critical heat flux ratio (MCHFR) will be 1.5 at design overpower (120% of rated power). The anticipated average burnup for the core will be 10,000 to 12,000 MWD/T. Unclad or vented fuels will not be used.

The fuel rod plenum pressure has been calculated by the applicant to be 1715 psia at the end of life at operating temperature. This pressure will not exceed the design stresses of the cladding for normal operation or transients resulting from malfunction of the reactor pressure controlling equipment.

The peak energy density within a fuel rod is about 60 cal/gram during normal operation and 130 cal/gram at design overpower. This energy density provides a sufficient design margin since clad damage and gaseous fission product release occur at about 170 cal/gram. The behavior of the fuel elements under the control rod dropout condition is discussed in Criterion 7.

We believe that this criterion is satisfied.

CRITERION 7

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted must be held to values such that no single credible mechanical or electrical control system malfunction could cause a reactivity transient capable of damaging the primary system or causing significant fuel failure.

The reactor control system contains 177 control rods having a total reactivity worth of 0.18. The rods can only be moved individually and the normal maximum reactivity addition rate is 0.0019 per second. The applicant has shown that accidental addition of reactivity at this rate as during a startup accident would not result in fuel damage.

Significant fuel damage could result from the dropout of a high worth control rod. The applicants' present calculations indicate that the maximum excursion that could be tolerated would be the dropout of a fuel rod worth 0.025 at a velocity of 5 ft/sec. This transient would result in a peak fuel enthalpy of 230 calories/gram. The UO_2 fuel melts in the range from 220-280 calories/gram and instantaneous clad failure and expulsion of molten and vaporized fuel need not be considered if the peak fuel enthalpy is less than 425 calories/gram. However, since there is some possibility that partial melting of the UO_2 within the fuel cladding due to such a reactivity transient could result in fuel movements which would add additional reactivity, General Electric plans to study the matter further during the detailed design of the reactor to insure that a maximum control rod worth of 0.025 is acceptable from a safety standpoint.

Procedural and engineered safeguards have been developed to assure that the maximum allowable control rod worth will not be exceeded during reactor operation. The withdrawal patterns of the control rods will be controlled by procedure so that no control rod has a reactivity worth greater than 0.025. Control rod w[illegible]hs of this magnitude can be achieved only d[illegible]ng low power operation; the maxim[illegible] control rod worth during power operation is about 0.01. In addition, a rod worth computer will be installed as a backup to the operating procedures. This device is a form of interlock which will prohibit the withdrawal of any control rod whose worth exceeded 0.025.

To limit the terminal velocity to 5 feet/second if a rod dropout should occur, a velocity limiter will be installed on each rod. The velocity limiter is a loose fitting piston which travels in the control rod guide tube, and is an integral part of the bottom of the control rod. The velocity limiter has no significant effect on the control rod insertion time under scram conditions. A thimble support will be installed which would limit the maximum downward movement of the thimble and attached rod to one or two inches thus preventing a control rod ejection should a control rod drive thimble fail.

We believe that the proposed design of Dresden Unit 2 satisfies this criterion.

CRITERION 8

Reactivity shutdown capability must be provided to make and hold the core subcritical from any credible operating condition with any one control element at its position of highest reactivity.

The maximum excess reactivity for the Dresden Unit 2 core in the cold clean condition is 0.26, and occurs at the beginning of life of the initial core. This reactivity excess is controlled by 177 control rods and 324 fixed control curtains worth 0.18 and 0.12, respectively. The reactivity worth for any single control rod removed from a cold clean core is less than 0.03, thereby assuring that the reactor can be held subcritical by 0.01 (k-effective = 0.99) for any credible condition with one rod withdrawn.

Future cores will be less reactive than the initial core since only a portion of the spent fuel elements will be replaced during any shutdown. The control curtains will be removed as required for future operation; but the capability to hold the reactor subcritical with one rod withdrawn will be maintained. Thus, the criterion is satisfied.

CRITERION 9

Backup reactivity shutdown capability must be provided that is independent of normal reactivity control provisions. This system must have the capability to shut down the reactor from any operating condition.

A standby liquid poison system has been provided to enable the reactor to be shut down safely if the control rods cannot be inserted. This system would insert sodium pentaborate solution in the reactor at the reactivity addition rate of -0.002 per minute, and has a total reactivity worth of at least 0.213. The negative reactivity insertion by this system is faster than the maximum reactivity gain that could be realized by cooldown of the moderator. Also, assuming that no control rod movement occurs, the system can maintain the reactor subcritical by at least 0.05.

We believe that this system satisfies the criterion.

CRITERION 10

Heat removal systems must be provided which are capable of accommodating core decay heat under all anticipated abnormal and credible accident conditions, such as isolation from the main condenser and complete or partial loss of primary coolant from the reactor.

If the reactor is either isolated from the main condenser or insufficient feedwater flow is available, the isolation condenser system will be used. Two identical isolation condensers will be provided. Each will consist of a 70 Mw capacity condenser which is 3% of the reactor power. The decay heat rate falls below 3% within 5 minutes. The condensers will be physically located above the reactor vessel in the reactor building. Operation will be by natural convection started by opening the line through which condensate would drain to the reactor vessel. The water in the shell side of the condensers would boil and the steam would be released to the atmosphere. Radiation monitors and provision for isolation will be available. Make-up water to the condensers will be from condensate storage tanks or, if necessary, the river via pumps on emergency power.

The equipment which would remove decay heat from the containment system under loss of coolant accident conditions consists of the two identical containment cooling loops which direct water from the suppression pool via pumps, a heat exchanger, and into the drywell via spray nozzles. The heat removal capacity of each loop is equivalent to that of the core decay heat rate. Operation of this equipment is initiated manually. The diesel generator is sized to operate this equipment, if required.

The relative importance under loss of coolant accident conditions of the containment cooling equipment discussed above and the core spray equipment discussed under Criterion 2 is illustrated by the fact that unless some containment cooling capacity is available, the integrity of the containment cannot be guaranteed beyond about 8 hours provided that no metal-water reaction occurs. If a metal-water reaction is assumed to occur, containment design pressure would be exceeded within about 1 hour of the accident in the absence of containment cooling.

In our opinion, the emergency cooling equipment discussed above is adequate in capacity and redundancy to satisfy the criterion.

CRITERION 11

Components of the primary coolant and containment systems must be designed and operated so that no substantial pressure or thermal stress will be imposed on the structural materials unless the temperatures are well above the nil-ductility temperatures. For ferritic materials of the coolant envelope and the containment, minimum temperatures are NDT + 60°F and NDT + 30°F, respectively.

The initial NDT temperature of the reactor vessel material opposite the core will be no higher than 10°F. The NDT of the remaining reactor vessel material will be no higher than 40°F. In the case of the Dresden Unit 2 vessel the presence of jet pumps in an annulus around the core provides a relatively large amount of water between the core and the vessel wall and thus a better capacity for thermalizing neutrons. Because of this the integrated exposure over the 40-year life of the reactor is estimated to be 5×10^{17}nvt. An exposure of 5×10^{17}nvt would not result in significant upward shift of NDT. Commonwealth states that the vessel will not be pressurized at a temperature below NDT + 60°F.

The containment steel is entirely enclosed within the reactor building and concrete structure and thus will not be exposed to the temperature extremes that would be experienced by a steel shell exposed to the environs. The containment environment temperature during normal operation will be 135°F. Charpy V-notch specimens of the containment steel will be tested to determine material properties.

We believe that these specifications satisfy the criterion.

CRITERION 12

Capability for control rod insertion under abnormal conditions must be provided.

The Dresden Unit 2 reactor will use G. E. type control rod drive mechanisms identical to those to be used in Jersey Central and Niagara Mohawk, and similar to those in use in Dresden Unit 1 and in the Big Rock Point reactor. The rods will scram upon failure of the pneumatic or electrical systems or upon failure of the scram pilot valve or inlet or outlet scram valves since the control valves are fail safe in this respect. When the reactor is at operating pressure, scram energy will normally be supplied from that which is stored in the scram accumulator. If required, the reactor pressure is also available for scram by movement of a ball check valve when the accumulator pressure falls below the reactor pressure.

The reactor vessel, all of its internals, as well as the control rods, drive system components and thimble supports will be considered Class I for seismic design purposes as described in Criterion 1. They will thus be designed to function throughout a severe seismic disturbance. These same components will be positioned well within the reinforced concrete structure and should thus be capable of withstanding effects of tornadoes. This is also discussed in Criterion 1.

Based on the foregoing, we believe that the control rod drive system satisfies this criterion.

CRITERION 13

The reactor facility must be provided with a control room from which all actions can be controlled or monitored as necessary to maintain safe operational status of the plant at all times. The control room must be provided with adequate protection to permit occupancy under the conditions described in Criterion 17 below, and with the means to shut down the plant and maintain it in a safe condition if such accident were to be experienced.

The Dresden Unit 2 facility will be equipped with a control room in which all controls and instrumentation necessary for operation of the reactor and turbine generator will be located.

With reference to occupancy during accidents, the applicant has stated that personnel in the control room will not receive doses in excess of 0.5 Rem in any 8 hour period following an accident involving either Unit 1 or Unit 2 and that the maximum potential doses for the course of an accident will not exceed 10 CFR 20 limits (3 Rem). These criteria can be readily met for Unit 2 which is buried below grade; however, Unit 1 is contained in a steel vessel above grade which would contribute significant radiation doses to personnel in the control room in the event of a core meltdown. The accident for Unit 1 for which the control rooms are designed is the "worst reasonable accident," a core meltdown and 25% $Zr-H_2O$ reaction. Direct radiation from the containment vessel, as well as air borne activity has been considered.

We believe that the foregoing satisfies the criterion. We note, however, that the applicant has not considered doses that would be received during access to and from the control rooms. We believe that this contribution should also be taken into account. The above, however, is a design consideration which can be satisfied during a detailed design of the facility.

CRITERION 14

Means must be included in the control room to show the relative reactivity status of the reactor such as position indication of mechanical rods or concentrations of chemical poisons.

Means for determining the relative reactivity status of the reactor will be provided by an individual readout for each rod displayed on a panel in the control room. Chemical poison would be used only under emergency conditions.

We believe that this criterion is satisfied.

CRITERION 15

A reliable reactor protection system must be provided to automatically initiate appropriate action to prevent safety limits from being exceeded. Capability must be provided for testing functional operability of the system and for determining that no component or circuit failure has occurred. For instruments and control systems in vital areas where the potential consequences of failure require redundancy, the redundant channels must be independent and must be capable of being tested to determine that they remain independent. Sufficient redundancy must be provided that failure or removal from service of a single component or channel will not inhibit necessary safety action when required. These criteria should, where applicable, be satisfied by the instrumentation associated with containment closure and isolation systems, afterheat removal and core cooling systems, systems to prevent cold-slug accidents, and other vital systems, as well as the reactor nuclear and process safety system.

Each sensor circuit (instrument channel) will be capable of being tripped independently by simulated signals for test purposes to verify its ability to give a single channel trip. Each sensor circuit, when tripped, deenegerizes its own individual relay, the contact(s) of which trip one and only one dual bus subchannel. Further, the tripping of a subchannel deenergizes only the scram logic relay associated with the subchannel. Thus, by successively observing the actions of the various combinations of two relays the operator can, without ambiguity, determine functional operability, and the presence or absence of circuit faults within the systems being tested. This includes, for example, short circuits which accidentally tie two or more dual bus subchannels together, thus destroying their independence. Testing, as described above, will reveal such a fault when a scram logic relay under test fails to deenergize.

Since the scram logic is 2-of-4 or 3-of-4, depending on which subchannels have tripped, it follows that the complete, unsafe failure of an entire subchannel will not preclude automatic scram by either the nuclear or process instrumentation. Further, each scram-producing parameter is monitored by not less than four independent sensor channels which, respectively, trip one and only one of the four dual bus subchannels. Thus, the complete unsafe failure of a sensor channel will not preclude automatic scram in response to an unsafe condition of its monitored parameter since such failure can, effectively, disable no more than one dual bus subchannel.

The applicant has stated that the failure of a single sensor circuit or system component will not prevent insertion of a sufficient number of control rods to shut down the reactor. From this we can infer that no single short-to-line at any group of parallel-connected pilot (solenoid) valves would prevent a safe shutdown.

The manual scram circuitry does not depend on the operation of any portion of the dual bus system; i.e. it is independent of the automatic scram system.

The foregoing discussion pertains to the protection system when no bypasses are in effect. Under such "non-bypass" conditions we have concluded that the criteria proposed by the applicant are in keeping with Criterion 15.

Under certain conditions of bypass, it has not been shown by the applicant that the portion of Criterion 15 which reads "Sufficient redundancy must be provided that failure or removal from service of a single component or channel will not inhibit necessary safety action when required" will be met. Specifically, the sensors of the eight intermediate range channels and eight power range channels are spatially distributed within each of four discrete quadrants in such a way as to protect the quadrants in the event of excessive flux occurring locally, or throughout the entire core. Within each range of instrumentation, two channels may be simultaneously bypassed provided they feed separate channels of the dual bus system, and are not protecting the same quadrant. This system of bypasses raises no difficulties concerning bulk power transients since these would be felt throughout the entire core and would be sensed by the unbypassed channels which exist in sufficient redundancy. However, a difficulty arises when local effects of single rod withdrawal are considered. Under the allowed conditions of bypassing, it is possible to have a region of a quadrant protected, in the vicinity of a local excursion-inducing rod, by a <u>single</u> channel of instrumentation capable <u>only</u> of providing a rod-blocking function. Should this single channel of protection fail, the consequences are similar to those resulting from a control rod dropout accident, i.e., severe damage to some 300 fuel rods.

The acceptability of this design approach is directly related to the probability that the protection system will not be in a failed state when it is called upon for protective action. This, in turn, is a function of the operating time during which bypasses are in effect and the frequency with which an active APRM channel is functionally tested during those intervals when it is a sole channel of protection.

If it can be shown that, by means of limited intervals of bypass and an increased frequency of testing during such intervals, the reliability of single channels of protective instrumentation can be made equal to the reliability of redundant channels of similar instrumentation, the applicant's design approach can be considered acceptable. Of some, though lesser, concern is the fact that intermediate range chambers may be bypassed such that two chambers, each protecting an adjacent quadrant, are widely separated, leaving a large portion of the core unmonitored and unprotected by this range of instrumentation. This circumstance is mitigated by the fact that the safety functions of the intermediate range channels are backed up by the APRM units.

Based on the discussion above, we believe that Criterion 15 can be satisfied since adequate procedural controls and testing during intervals of bypass can be provided.

CRITERION 16

The vital instrumentation systems of Criterion 15 must be designed so that no credible combination of circumstances can intefere with the performance of safety function when it is needed. In particular, the effect of influences common to redundant channels which are intended to be independent must not negate the operability of a safety system. The effects of gross disconnection of the system, loss of energy (electric power, instrument air), and adverse environment (heat from loss of instrument cooling, extreme cold, fire, steam, water, etc.) must cause the system to go into its safest state (fail-safe) or be demonstrably tolerable on some other basis.

The individual subchannels within the dual bus scram system, and the pilot (solenoid) valves are fail-safe with respect to loss of voltage. The scram valves are fail-safe with respect to loss of instrument air.

Further, the applicant has stated that each scram trip will be designed "fail safe" insofar as practicable, i.e. most probable component failures (including power supply failure) or open circuit in wiring shall cause a trip condition. Specifically, line failures at active startup, intermediate or power range nuclear instrumentation feeding the scram system will always actuate a "downscale" rod block interlock. In most cases the relays which trip the dual bus subchannels will also deenergize (fail safe).

Process system scram contacts are connected directly into the dual bus system; i.e., there are no intermediate relays, amplifiers, etc. Thus, they are independent of power sources. "Fail safety" can only be discussed in terms of mechanical operation and, in our opinion, a switch can fail in either mode with equal probability. There is, however, as stated herein, redundancy within each process sensor system, and we believe that this redundancy fulfills the "demonstrably tolerable on some other basis" portion of Criterion 16.

We have concluded that the design criteria proposed by the applicant satisfies Criterion 16.

ENGINEERED SAFEGUARDS

CRITERION 17

The containment structure, including access openings and penetrations, must be designed and fabricated to accommodate or dissipate without failure the pressures and temperatures associated with the largest credible energy release including the effects of credible metal-water or other chemical reactions uninhibited by active quenching systems. If part of the primary coolant system is outside the primary reactor containment, appropriate safeguards must be provided for that part if necessary, to protect the health and safety of the public, in case of an accidental rupture in that part of the system. The appropriateness of safeguards such as isolation valves, additional containment, etc., will depend on environmental and population conditions surrounding the site.

Commonwealth Edison quotes a similar criterion for the design of the containment structure as follows: "To withstand the peak transient pressure which could occur due to the postulated rupture of any reactor system primary pipe inside the drywell."

The design pressure of both the drywell and the suppression pool is 62 psig coincident with a temperature of 281°F. The pressures and temperatures associated with a loss of coolant accident, rupture of a 28-inch diameter recirculation loop, have been calculated both with and without the presence of Zr-H_2O reaction, and assuming that various combinations of engineered safeguards do not function. The information presented indicates that there is a large margin between the accident overpressure and temperature and the design pressure and temperature of 62 psig and 281°F, respectively, provided that at least one of the suppression pool cooling-containment spray loops remain in operation and the containment system is inerted to preclude recombination of hydrogen from a Zr-H_2O reaction. The peak drywell pressure in the early stages of the accident before significant Zr-H_2O reaction can take place is shown to be 38 psig. This pressure is based upon an analytical model developed from results of Moss Landing tests conducted by the Pacific Gas & Electric Company. Subsequently the pressure falls below this value, but remains at about 20 psig for many hours due to decay heat and Zr-H_2O reaction. The containment temperature after the loss of coolant reaches just above 200°F for a short period of time. Subsequently it falls to lower values. This assumes, as before, one suppression pool cooling-containment spray loop in operation.

Four main steam lines penetrate the containment system and may be considered an extension of the primary coolant system. Protection against failure of these lines is provided by two isolation valves in series, one on each side of the containment vessel wall, in each of the four lines. These valves are designed to close within 11 seconds after a steam line failure. Since available information does not preclude the possibility that a single failure at or near one of the isolation valves, or inthe section of piping between the valves, could lead to sequential failure of both valves, we believe this potential problem should be given special attention during the detailed design of the facility in order to eliminate the possibility of such an occurrence. However, each of the steam lines is also equipped with a steam flow restrictor which would choke steam flow in a given line to twice normal flow if a failure were to occur, and thus limit uncontrolled release of steam. The restrictors are placed as close as possible to the reactor vessel within the containment.

The two isolation condensers communicate directly with the primary coolant system and may also be considered an extension of the primary coolant system. The lines to these condensers are equipped with two isolation valves in series, one on each side of the pressure vessel wall, which close upon indication of an isolation condenser failure.

In the case of Dresden Unit 2, taking into consideration the low population density near the site, we believe that the redundancy of isolation valves provided is adequate to protect the health and safety of the public.

We believe that the Dresden Unit 2 proposed containment design satisfies this criterion.

CRITERION 18

Provisions must be made for the removal of heat from within the containment structure as necessary to maintain the integrity of the structure under the conditions described in Criterion 17 above. If engineered safeguards are needed to prevent containment vessel failure due to heat released under such conditions, at least two independent systems must be provided, preferably of different principles. Backup equipment (e.g., water and power systems) to such engineered safeguards must also be redundant.

The equipment supplied to remove decay heat from the containment system during accident conditions consists of two identical containment cooling loops which direct water from the suppression pool via pumps to a heat exchanger and into the drywell via spray nozzles. The heat removal capacity of each loop is equivalent to that of the core decay heat rate. The containment system pressures and temperatures that would result assuming operation of this one loop are discussed under Criterion 17.

The heat sink for the containment cooling heat exchangers will be river water. Water from the cooling water intake will be pumped directly to the containment cooling heat exchangers by at least two full capacity pumps.

Electrical power for this equipment will be available from any one of the power sources entering the facility or if these fail, the emergency diesel generator. (See also Criterion 21).

Based on these considerations, we believe this criterion is satisfied.

CRITERION 19

The maximum integrated leakage from the containment structure under the conditions described in Criterion 17 above must meet the site exposure criteria set forth in 10 CFR 100. The containment structure must be designed so that the containment can be leak tested at least to design pressure conditions after completion and installation of all penetrations, and the leakage rate measured over a suitable period to verify its conformance with required performance. The plant must be designed for later tests at suitable pressures.

A similar design criterion for the containment system has been quoted by Commonwealth as follows:

"To limit primary containment leakage during the following a postulated rupture of the reactor primary system to a value which is substantially less than the leakage rates which would result in off-site doses approaching the reference dose in 10 CFR 100," and

"To include in the design provisions for periodic leakage tests."

That leakage rate which has been specified to satisfy the site exposure criteria set forth in 10 CFR 100 is 0.5% per day. This is discussed in the "Accident Analysis" section of this report in which it is shown that containment leakage at this rate would result in off-site exposures significantly less than those stated in 10 CFR 100. We believe that a leakage rate of 0.5% per day can be achieved readily by careful design and fabrication of the containment system and that a leakage rate of this magnitude is amenable to verification by test.

As noted previously, the design pressure of both the drywell and suppression pool is 62 psig. After installation of all penetrations, the applicant intends to demonstrate by tests that the integrated leakage rate from the vessels does not exceed 0.5% per day at 62 psig. In addition, the applicant intends to measure leakage rate at different lower pressures to obtain a leakage characteristic curve of the containment system. Using this curve for extrapolation of leakage to higher pressures, subsequent integral leakage tests will be conducted at lower than design pressures. The pressure at which subsequent tests will be conducted and the leakage rate to be allowed will be decided during operating license procedures after the initial tests have been conducted.

We believe that design of the containment, the specified gaseous leakage rate, and the proposed tests are adequate to meet this criterion.

It appears that leakage from suppression pool coolers under accident conditions could also contribute significantly to overall leakage of fission products to the environment. Neglecting leakage from other components in these loops, the applicant calculated that under major loss of coolant accident conditions, the release of fission products via leaking suppression pool water from the suppression pool cooling loop pumps would be roughly an order of magnitude less than the leakage of fission products in gas from the drywell and suppression pool. Nevertheless, we believe that the importance of this source of leakage of fission products should not be discounted, since inadvertent leakage from a pump or other component in the loop could contribute significantly to the overall off-site dose. For this reason, we believe that a maximum leakage from this equipment should be specified and that means and a program should be available to leak test this equipment. We believe such a program can be developed and the criterion can be satisfied in this respect.

CRITERION 20

All containment structure penetrations subject to failure such as resilient seals and expansion bellows must be designed and constructed so that leak-tightness can be demonstrated at design pressure at any time throughout operating life of the reactor.

The following leak detection capability for individual penetrations will be provided:

1. Leak detection tests at 100% design pressure of all penetrations fitted with resilient seals or gaskets.
2. Leak detection testing to at least 100% design pressure and operability tests under pressurized containment conditions, of the isolation valves of (a) systems open to the containment, and (b) systems whose pipelines connect to the reactor system.
3. Leakage and operability testing of the engineered safeguard systems directly connected to the containment vessels under containment pressurized conditions.

We believe that this leak detection capability satisfies the criterion.

CRITERION 21

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

The Dresden Unit 2 facility is well supplied with auxiliary and emergency sources of electrical power. There are available four separate and independent sources of power. They are:

1. The Unit 2 generator,
2. The 138 KV transmission system,
3. The standby diesel generator, and
4. The station battery (125 VDC).

By the time Unit 2 is completed, there will be six 138 KV transmission lines serving the site. Five will enter on a common right-of-way from the north, the sixth will enter from the south. In addition, Unit 1 feeds the 138 KV switchboard.

A 34.5 KV line will enter the plant from the south on a right-of-way which will be underground for about 1100 feet prior to entering the plant area.

The diesel generator will be situated in a reinforced concrete block cell in the turbine building. This generator will be placed in standby operation whenever there is a tornado alert which would be likely reason for loss of off-site power. It will be of sufficient size to operate all electrical equipment necessary to protect the Unit during a normal power outage as well as a power outage concurrent with an accident.

All electrical equipment needed to protect the Unit is protected from tornado winds by underground cable and metal enclosed switchgear within the turbine building.

We believe that Dresden Unit 2 is well protected from simultaneous loss of all electrical power which would hinder a safe plant shutdown and that this criterion is satisfied.

CRITERION 22

Valves and their associated apparatus that are essential to the containment function must be redundant and so arranged that no credible combination of circumstances can interfere with their necessary functioning. Such redundant valves and associated apparatus must be independent of each other. Capability must be provided for testing functional operability of these valves and associated equipment to determine that no failure has occurred and that leakage is within acceptable limits. Redundant valves and auxiliaries must be independent. Containment closure valves must be actuated by instrumentation, control circuits and energy sources which satisfy Criterion 15 and 16 above.

The criteria quoted by the applicant for the installation of isolation valves are as follows:

1. Process pipes which connect to the reactor primary system, and pipes or ducts which penetrate the primary containment and are open to the drywell free air space shall be provided with at least two isolation valves in series.

Valves in this category shall be designed to close automatically from selected signals, and shall be capable of remote manual actuation from the control room.

2. The valves will be physically separated. On lines connecting to the reactor primary system, one valve shall be located inside the primary containment and the second outside the primary containment as close to the primary containment wall as practical.

3. Lines which penetrate the primary containment and which neither connect to the reactor primary system nor which open into the primary containment, shall be provided with at least one valve which may be located outside the primary containment.

Valves in this category shall be capable of manual actuation from the control room.

4. Automatic isolation valves, in the usual sense, will not be used on the inlet lines of the core spray, core reflooding, containment spray, and feedwater water systems, since operation of these systems is essential following a loss-of-coolant accident. Since the normal flow of water in these systems is inward to the reactor vessel or primary containment, check valves located in these lines, inside the drywell, will provide automatic isolation when necessary. The check valves are powered by reverse (outward) fluid flow.

5. Automatic isolation valves will not be provided on the outlet lines from the pressure suppression chamber to the core cooling and containment cooling pumps. These lines return to the containment and are required to be open during post accident conditions for operation of these systems.

6. No automatic isolation valves are provided on the control rod drive hydraulic system lines. These lines are isolated by means of the normally closed hydraulic system control valves located in the reactor building, and by means of check valves comprising a part of the drive mechanism.

7. Small diameter instrument lines are provided with one manually operated shut-off valve, operable from the reactor building.

8, Motive power for the valves on process lines which require two valves shall be physically independent sources to provide a high probability that no single accidental event could interrupt motive power to both closure devices.

9. Upon loss of motive power and when containment closure action of

the valve is called for, the valve shall fail closed or shall fail in it existing position.

10. Valve actuation power failure shall be detected and annunciate

11. Isolation valve closure time shall be such that for any design basis break the coolant loss is restricted to an amount less than that would result in uncovering the core.

12. Valves, sensors, and other automatic devices essential to the isolation of the containment shall be provided with means to periodical test the functional performance of the equipment. Such tests would inc demonstration of proper working conditions, correct set point of senso proper speed of responses, and operability of fail safe features.

We believe that these criteria stated by the applicant satisfy th criterion. The leakage testing capability is discussed under Criterion 20.

There is one exception to the above. The guide tubes for the Traversing Incore Probe (TIP) calibration system run from the reactor core, through the containment, to the reactor building. The Telflex type cables will be used to drive the probe. When the system is in use, the isolation valve on the guide tube will be required to be open. Although one isolation valve on each guide tube has been specified, G. E. is now devising a means by which the equivalent of two isolation valves in series can be provided. In this manner the TIP system lines would have isolation capability at least equivalent to the performance objective in the Plant Design and Analysis Report which states:

"Process pipes which connect to the reactor primary system, and pipes or ducts which penetrate the primary containment and are open to the drywell free air space shall be provided with at least two isolation valves in series. Valves in this category shall be designed to close automatically from selected signals, and shall be capable of remote manual actuation from the control room."

In view of the solution proposed by General Electric, we believe this criterion will be satisfied in this respect.

CRITERION 23

In determining the suitability of a facility for a proposed site under Part 100, the acceptance of the inherent and engineered safety afforded by the systems, materials and components, and the associated engineered safeguards built into the facility, will depend on their demonstrated performance capability and reliability and the extent to which the operability of such systems, materials, components, and engineered safeguards can and needs to be tested and inspected during the life of the plant.

Each of the principal engineered safeguards and the manner in which they are to be tested is listed below:

1. Core spray systems. Tentatively, these systems are capable of testing by circulating water by the pumps up to the final motor operated block vlave. This valve can be tested separately; however, the details of periodic testing have not been finally established. Commonwealth is considering ways in which the spray rings can be tested. A preoperation functional test of the entire system is planned. The amount of water required to cool an individual fuel assembly has been measured by G.E. during tests on simulated dry fuel elements. Approximately 3 gpm per fuel assembly will be required.

2. Containment of cooling systems. Tentatively, these systems are capable of testing by circulating water by the pumps up to the final motor operated block. This valve can be tested separately; however, the detai

of periodic testing have not been finally established. Commonwealth is considering ways in which the spray nozzles can be tested. A preoperational functional test of the entire system is planned.

3. Reactor Building. The leak tightness of the reactor building can be tested by isolation and exhausting air via the standby gas treatment system. The rate at which air is exhausted at the design pressure of 0.25-inch of water negative pressure is a measure of the in-leakage.

4. Standby gas treatment system. This system will be designed so that periodic tests of fans and controls, the differential pressure across each filter, and the filter efficiency can be measured and tested.

5. Core reflooding. This capability will be tested during preoperational checkout of the Unit. The nature of the safeguard precludes subsequent tests.

6. Control velocity limiter. Out-of-core development testing has been performed. The nature of this safeguard precludes functional testing of the installed equipment.

7. Steam line flow restrictor. Laboratory development tests have been made. As above, the nature of this safeguard precludes functional testing of the installed equipment.

8. Control rod drive thimble support. This structural equipment is not amendable to functional testing.

We believe that the testing capability to be provided for the eight engineered safeguards listed above satisfies the criterion.

RADIOACTIVITY CONTROL

CRITERION 24

All fuel storage and waste handling systems must be contained if necessary to prevent the accidental release of radioactivity in amounts which could affect the health and safety of the public.

The spent fuel storage pool and the new fuel storage vault are located inside the reactor building which provides confinement of radioactive materials which would be released during a refueling accident. The consequences of this accident are discussed in the Accident Analysis section of this report. We believe this criterion is satisfied.

CRITERION 25

The fuel handling and storage facilities must be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel under all anticipated normal and abnormal conditions, and credible accident conditions. Variables upon which health and safety of the public depend must be monitored.

In the refueling procedure, all operations are carried out with the fuel under water which enables visual control of operations at all times. Reactor spent fuel is transferred under water through a spent fuel storage pool canal into racks provided in the storage pool. The storage pool is designed to accommodate all the required fuel maintenance operations and storage space is also provided in the pool for the control rods, fuel shipping cask, and small internal reactor components. Ample storage space is provided in the event of a complete core unloading. Additional storage for large components is provided in a separate storage pool adjacent to the drywell head cavity.

In order to avoid accidental draining of the fuel storage pool, there are no penetrations that would permit the pool to be drained below a safe storage level and all lines extending below this level are equipped with suitable valving to prevent backflow. The passage between the fuel storage pool and the refueling cavity above the reactor vessel is provided with two double-sealed gates with a monitored drain between the gates permitting detection of leaks from the passage and repair of a gate in the event of such leakage.

The pool system also contains provisions to maintain water cleanliness and instrumentation to monitor water level. The rack in which fuel assemblies are placed is designed and arranged to ensure subcriticality in the pool.

Water depth in the pool will be such as to provide sufficient shielding for normal occupancy of operating personnel. In addition, the fuel storage pool is a reinforced concrete structure completely lined with seam-welded stainless steel plates welded to reinforcing members (channels, I-beams, etc.) embedded in concrete. The pool will be designed to adequately withstand the anticipated earthquake loadings as a Class I structure. The lower liner section will be built as a free standing structure and hydrostatically tested prior to the concrete pour. The stainless steel liner should prevent leakage even in the event of cracks developing in the concrete.

New fuel is brought into the reactor building through the equipment entrance and hoisted to the upper floor utilizing the reactor building crane. The new fuel is stored in the reactor building in the new fuel dry vault located convenient to the refueling pool area and serviced by work area equipment.

The storage racks for new fuel are full length, top entry, and designed to prevent an accidental critical array even if the vault becomes flooded. Vault drainage is provided to prevent possible water collection, and all entrances to the vault are capable of being locked.

Thus, we believe that this criterion is satisfied.

CRITERION 26

Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate hold-up capacity must be provided for retention of gaseous, liquid, or solid effluents.

The process off-gas system handles radioactive gases of plant origin. Noncondensable radioactive gases are removed from the main condenser by the air ejector which exhausts these off-gases into shielded piping providing a 30-minute holdup for radioactive decay of short-lived isotopes. These gases are then passed through a set of two high efficiency particulate filters prior to release to a 300-foot stack. A spare set of filters is also provided to assure availability of filtration. Off-gas monitors will be provided which will automatically shut the isolation valves in the off-gas line should radiation levels above some pre-set value occur.

The liquid radioactive waste system collects, treats, stores, and disposes of all radioactive liquid wastes. These are collected in sumps and drain tanks and then transferred to the appropriate tanks in the Radwaste Building for further treatment, storage, and disposal. Wastes to be discharged from the system are handled on a batch basis with each batch being analyzed and handled appropriately. Final disposition of processed liquid wastes consists of return to the condensate system, storage awaiting solidification and disposition off-site as solid wastes or disposal through the discharge canal. Those batches whose radioactivy concentrations are sufficiently low as to allow disposal in the Illinois River are released into the discharge canal and diluted with effluent condenser circulating water in order to achieve a discharge concentration of 10^{-7} uc/cc at the point of entry into the river.

The principal origins of solid radioactive wastes are those from the reactor, maintenance of equipment, and operation of the process systems. The reactor wastes are stored for decay in the fuel storage pool, packaged and transferred to permanent disposal off-site in shipping containers. The maintenance wastes are compressed into bales to reduce volume and packaged for disposal. The process wastes are collected in tanks, dewatered, drummed in 55-gallon containers, and stored awaiting shipment. Concentrated wastes are mixed with water absorbent material prior to drumming and subsequently stored awaiting disposal.

All of the radioactive waste disposal systems appear to contain appropriate holdup capacity for radioactive decay or provision for adequate dilution. Instrumentation is also provided for detection and alarm of abnormal conditions, and thus we believe that this criterion is satisfied.

CRITERION 27

The plant must be provided with systems capable of monitoring the release of radioactivity under accident conditions.

Excluding exfiltration from the reactor building, all of the radioactive effluent from an accident will be discharged by the facility stack which is monitored. To supplement the stack monitor, portable survey equipment will be kept in the control room. The particular procedures and equipment to be used in the case of an accident have not as yet been specified. However, the applicant has stated that they will generally follow the procedures now used in Unit 1. This includes the control room personnel notifying the utility load dispatcher who in turn would notify local authority and Argonne National Laboratory.

In view of the foregoing, we believe that this criterion is satisfied.

VI. Accident Analysis

In the Plant Design and Analysis Report, Commonwealth described the course and consequences of a number of relatively minor accidents, including control rod drive system malfunction or maloperation, cold water addition accidents, malfunction of coolant circulation equipment, steam valve failures, and minor loss of coolant accidents. None of these accidents are considered to be serious in terms of off-site consequences since they would not result in failure of the fuel cladding and release of fission products if it is assumed the normal safety systems function as designed. In addition, Commonwealth has described in some detail the consequences of a control rod drop accident, a fuel loading accident, a steam line rupture (outside containment) accident, and a major loss of coolant accident. The analysis of the consequences of these accidents demonstrate that if engineered safeguards function and other attenuating factors which would tend to mitigate the accident consequences occur as anticipated, it is difficult to postulate an accident which would result in a significant off-site health hazard. In this manner, the applicant has presented the probable consequences of credible accidents to the facility.

Our analysis given in the following sections, provides an evaluation of the possible maximum consequences of these same serious accidents. The Commission's Site Criteria (10 CFR 100) state that potential radiation doses received at the exclusion area boundary during the first two hours following a credible but highly unlikely accident should not exceed 300 Rem to the thyroid or 25 Rem whole body. Under the site criteria, these same doses should not be exceeded at the outer edge of the low population zone, in this case 10 miles, during the course of the accident. Our evaluation indicates that the doses would be within those stated in 10 CFR 100, assumming the functioning of some but not all of the engineered safeguards to be provided. The doses tabulated below are those which could be received off-site for the first two hours after the accident at the 1/2 mile exclusion area boundary, and for the course of the accident at the 10 mile low population zone boundary. For the refueling and loss of coolant accidents, a Standby Gas Treatment System halogen filter efficiency of 90% was assumed.

TABLE I

Accident	POTENTIAL OFF SITE DOSES (REM)					
	First Two Hours at 1/2 mile		Worst Two Hours at 1/2 mile		Course of Accident 10 miles	
	Thyroid	Whole Body	Thyroid	Whole Body	Thyroid	Whole Body
Steam Line Break	10	--	--	--	<1	--
Control Rod Drop	100	10	--	--	20	2
Refueling*	150	4	--	--	30	<1
Loss of Coolant*	<1	<1	5	<1	150	6

* Standby Gas Treatment System halogen filter efficiency of 90% assumed.

The assumptions used in determining these consequences are discussed in the following sections. The reason for presenting the doses received during the worst two hours from the loss of coolant accident is that these doses are significantly higher than for the first two hours, which is the time period referred to in 10 CFR 100. The first two hours potential exposure are less severe because leakage from the containment system is first into the reactor building and some time is required to achieve equilibrium concentration of fission products in this building.

A. Steam Line Break

This accident results from a circumferential rupture in the pipe tunnel leading to the turbine of one of the 20-inch main steam lines. The isolation valves close in 11 seconds and the flow restrictors choke flow to twice normal flow. It is also assumed that (1) the water and steam blowdown total 70,000 lb during the 11 seconds before isolation valve closure, (2) 100% of the halogens contained in the water and steam remain airborne, and (3) the fission product inventory released is equivalent to that in 70,000 lb of reactor water when the stack release rate is at the license limit. This inventory is at least an order of magnitude higher than that expected during normal operation. With these assumptions, about 100 curies of I-131 and I-133 would be released at ground level in a cloud of steam. TID meteorology was assumed and the steam cloud was assumed not to be buoyant. The dose contribution from noble gases is negligible since they are not absorbed in the reactor water but are removed continuously by the off-gas system. Consequences would be as given in Table I.

B. Control Rod Drop

The accident involves a control rod falling from the core when the reactor is at hot standby, the "worst case". The control rod is worth 0.025 which is the highest reactivity worth permitted by the rod worth limiter. The drop velocity is limited to 5 ft/sec by the rod velocity limiter.

It is assumed that at hot standby the fission product decay heat is being dissipated to the main condenser and the condenser vacuum is maintained by the mechanical vacuum pump which discharges to the stack via a 1.75 minute holdup line. This accident is significant because the Standby Gas Treatment System, one of the engineered safeguards could be bypassed and the effluent passed directly to the facility stack.

This accident would result in a power transient in which an estimated 330 fuel rods would suffer clad perforation (fuel enthalpy above 170 cal/gm) but no fuel would melt (fuel enthalpy above 220 cal/gm). We assumed that (1) 50% of the halogens and 100% of the noble gases contained in the perforated fuel rods are released to the reactor water, (2) 10% of the released halogens are carried to the condenser hotwell, and (3) 50% of the remaining halogens plate out in the condenser. The remaining radioactivity would be discharged via the stack and consequences would be as given in Table I.

C. Refueling Accident

The refueling accident is the postulated drop of a fuel element into a near critical fuel array obtained by withdrawing two adjacent control rods during refueling. The accident could only occur with the violation of a number of procedural requirements plus the failure of interlocks which prevent fuel handling over the reactor while control rods are withdrawn. The fuel assembly reactivity worth in the postulated "worst case" configuration is 0.023 and the reactivity insertion rate is 0.10 per second.

The applicant has calculated, and we agree, that the excursion would generate about 2610 Mw-sec of energy and that 224 fuel rods would be damaged. This is in general agreement with SPERT oxide core tests. No fuel would be melted. We assumed that 50% of the halogens and 100% of the noble gases in the perforated fuel are released to the water and that 50% of the released halogens remain in the water or plate out on cold surfaces. The remaining radioactivity would be released via the Standby Gas Treatment System and the stack. Consequences would be as given in Table I.

D. Loss of Coolant Accident

The break of a pipe in the primary system within the drywell is considered the Maximum Credible Accident. A break in the "second line of defense" against fission product release, the primary system, could lead to violation of the "first line of defense"; the fuel clad. This places complete reliance on the last major barrier to fission product release; the drywell-suppression chamber containment.

A range of coolant loss accidents has been analyzed, and the largest break, a circumferential rupture of the 28-inch diameter recirculation line, results in the highest containment pressure. Two engineered safeguards must function to prevent major fuel melting; (1) one of the two core spray systems and (2) one of two containment cooling system loops in which water is taken from the suppression pool, passed through a heat exchanger, sprayed into the drywell and returned to the suppression pool via the vent lines.

After the break in the 28-inch recirculation line the primary system steam and water would blow down within about 20 seconds to the containment system. Peak pressures during the blowdown are calculated to be 43 psig in the drywell and 21 psig in the suppression chamber as opposed to the design pressure of 62 pisg for each chamber. The containment pressures are reduced to about 3 psig in the first few minutes due to the steam condensation effected by the drywell spray. The core spray, initiated by low primary system pressure, would come on within one minute following the break. This spray would prevent the UO_2 from reaching 3000°F, the re-crystalization temperature, if initiated as late as 5 minutes after the break. For the case in which the core spray functions properly, about 1% of the zirconium in the core would react and the evolved hydrogen would not give rise to a flammable mixture within the containment. Off-site radiation doses would be significantly lower than those given in Table I for the 100% core meltdown case.

It is our opinion that while all possible steps should be taken to prevent major fuel melting during a coolant loss accident, the design basis of the tainment should be the containment of the fission products from a 100% core iown with attendant metal-water reaction. In our view, the applicant has strated that the proposed containment system meets this design basis. e unlikely event that the core spray fails, the core would heat up and zirconium clad and fuel channel material would react with any residual er left in the reactor vessel. The applicant has calculated that not more an a 27.5% reaction of the zirconium would result. The applicant has also calculated that burning the hydrogen from only a 4% zirconium-water reaction would give rise to a containment pressure level of 62 psig. Accordingly, the applicant proposes to provide an inert containment atmosphere to avoid recombination of the hydrogen and the resulting excessive pressures.

The extent of metal-water reaction was calculated by assuming that the core heated up with no cooling after the blowdown, that the metal-water reaction followed an accepted reaction rate as a function of local metal temperature and the local extent of the reaction, that the reaction was supplied with the stoichiometric amount of steam at each point of the core, and that the reaction terminated when the local clad temperature reached 3300°F, the melting temperature of zirconium. The reaction in the core was calculated to total 24.5% of the core zirconium and another 3% was assumed to react as the molten core dropped into the lower plenum. The 24.5% reaction was estimated to take about 20 minutes. It is our opinion that the short time calculated for the completion of the metal-water reaction is conservative.

The applicant has presented a calculation that illustrates that the containment system could withstand even higher zirconium-water reactions extended over a longer period of time; for example, 70% in 3 hours. It is assumed that only one of the two containment cooling loops functions. Thus, although there could be some question regarding the point of termination of the zirconium-water reaction, since there is evidence that the integrity of the clad may be maintained by the ZrO_2 formed, we believe that the calculated time rate of reaction is conservative since the reaction would be limited to a slower rate by the rate of steam evolution from the bottom plenum. With this reasoning we believe that it is logical to accept the applicant's calculations of zirconium-water reaction, recognizing the limitations on the calculation techniques, that 27.5% in 20 minutes is conservative, and that a significantly higher reaction percentage could be tolerated over an extended period of time.

The off-site doses which would ensue as a consequence of this accident are given in Table I. The following assumptions were made in the staff calculation: (1) 50% halogen and 100% noble gas release from the fuel,

(2) 50% plateout of halogens in the drywell, (3) 0.5% per day leak rate to the reactor building, and (4) 100% per day of the reactor building volume released to the stack via the Standby Gas Treatment System.

The above illustrates that some credit must be assigned to the engineered safeguards to bring the potential consequences of the accidents to within those suggested in 10 CFR 100. Although Table I indicates the requirement of a halogen filter efficiency of 90% in the Standby Gas Treatment System, the core spray is also available to limit the core meltdown which would decrease the fission product source. We believe that, in combination, these safeguards as well as other factors that would reduce the source of fission products will provide the necessary reduction in off-site exposure due to this accident.

VII. Research & Development

On all components which are important for the safe operation of the Unit, the architectural and engineering criteria have been described. At this stage in design, the applicant has not completed the final layout arrangements and design details of many components and systems of the plant. Programs are being conducted which will aid in determination and evaluation of the final design. These include:

1. Development of in-core instrumentation,
2. Development of control rod worth minimizer,
3. Development of a rod dropout velocity limiter,
4. Development of jet pumps, and
5. Development of control rod th [illegible]e support.

Our evaluation of the information submitted thus far leads us to believe that acceptable design details can be evolved from the programs proposed. At a later stage of development, a description of the final design derived on the basis of these programs will be submitted by the applicant and will be evaluated by the Staff.

VIII. Technical Qualifications

The applicant, Commonwealth Edison Company, has been operating Dresden Unit 1 on the Dresden site for about five years with considerable success. The administrative procedures and personnel training which have proven to be successful with Unit 1 will be applied to Unit 2. We believe that there is no better way to demonstrate technical competence in the reactor operation field than to have had operating experience with a reactor similar to the one which is planned. We thus have no reservations concerning the technical competence of Commonwealth.

The nuclear contractor, the General Electric Company, has been engaged for a number of years in design, construction, and operation of boiling water reactors. Among the operating reactors with which the General Electric Company has been associated are the Vallecitos, Dresden, Humboldt Bay, Big Rock Point, and SENN reactors. They have also designed the recently licensed Jersey Central and Nine Mile Point facilities which are boiling water reactors essentially similar to Dresden Unit 2. Thus, General Electric has a substantial knowledge and capability with the type of reactor proposed and is capable of discharging its responsibilities to the applicant in this facility.

With this staff and experience, and based upon our evaluation of the personnel responsible, we believe that the applicant is suitably qualified to design and construct the proposed facility.

IX. Report of the Advisory Committee on Reactor Safeguards

A subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) met with representatives of Commonwealth on September 1, 1965 to consider the proposed Unit 2. Subsequently, during its sixty-seventh meeting in October, 1965 and during its sixty-eighth meeting in November, 1965, the full ACRS reviewed this application and discussed the proposed Unit with representatives of Commonwealth. A copy of the ACRS letter to the Commission concerning the Commonwealth Edison application for a construction permit for Dresden Unit 2 is attached as Appendix A.

The ACRS in this letter included several recommendations concerning the design of the proposed facility, but concluded that these problems can be resolved during development of the final design. We have considered each of these matters and agree with the ACRS that they can be resolved during construction. The letter then concluded ". . . that the Dresden 2 reactor can be built and operated at the proposed site without undue risk to the health and safety of the public."

X. Conclusions

Based on the proposed design of the Dresden Nuclear Power Station, Unit 2, on the criteria, principles and design arrangements for systems and components thus far described, which includes all of the important items, on the calculated potential consequences of routine and accidental release of radioactive materials to the environs, on the scope of the development program which will be conducted, and on the technical competence of the applicant and the principal contractor which will design and construct the plant, we have concluded that, in accordance with the provisions of paragraph 50.35(a), 10 CFR 50:

1. The applicant has described the proposed design of the facility, including the principal architectural and engineering criteria for the design, and has identified the major features or components on which further technical information is required;

2. The omitted technical information will be supplied;

3. Research and development as required to resolve the safety questions with respect to the features and components which require research and development will be conducted;

4. On the basis of the foregoing, there is reasonable assurance that (1) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for the completion of construction of the proposed facility, and (2) taking into consideration the site criteria contained in Part 100 of the Commission's regulations, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public;

5. The applicant and its contrac[illegible] are technically qualified to design and construct the proposed facility;

6. The issuance of a provisional construction permit for the proposed facility will not be inimical to the common defense and security or the health and safety of the public.

In summary, we have concluded that there is reasonable assurance that the Dresden Nuclear Power Station, Unit 2 can be constructed and operated at the proposed site without endangering the health and safety of the public.

Appendix A

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

NOV 24 1965

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON DRESDEN NUCLEAR POWER STATION - UNIT 2

Dear Dr. Seaborg:

At its sixty-seventh and sixty-eighth meetings, the Advisory Committee on Reactor Safeguards considered the Dresden 2 reactor which the Commonwealth Edison Company proposes to build on the Dresden site. The applicant proposes a boiling water reactor with pressure suppression containment designed by the General Electric Company. The design power level, 2255 MW(t), is about three times that of Dresden 1. The Committee had the benefit of discussions with representatives of the applicant, the General Electric Company, Sargent & Lundy, and the AEC Staff and its consultants, and of the documents listed. A Subcommittee of the ACRS met to review this project on September 1, 1965.

At the sixty-eighth meeting, an extensive development program was described by the representatives of the General Electric Company. The results of these mockup experiments and calculations are still preliminary. The program is intended to answer questions on the following: jet pump monitoring and system stability, metal-water reactions, instrumentation, and blow-down and emergency cooling. The Committee recommends continued studies of pipe-whipping and the generation of missiles which might violate the containment during a postulated accident involving failure of the primary system piping. The possible effect of fuel movement upon reactivity transients will also be considered by General Electric. The Committee recommends that the AEC Staff follow the results of these programs closely.

The Committee urges that the designers of the pressure vessel be especially attentive to problems which may arise due to its large size. In particular, the Committee would like to learn if any effects such as bellmouthing or vibration become important as vessel sizes increase. In addition, the Committee recommends that a careful check be made of the vessel designer's Stress Report for the applicant by General Electric or by independent experts. The Committee also urges that efforts be made to provide means of access for periodic inspection of the reactor vessel for flaws.

While the isolation valves and fittings in the high pressure steam lines from the reactor to the turbine are being designed carefully, the Committee recommends that special attention be given to insure that no single rupture can lead to sequential failure and loss of containment.

It is the opinion of the ACRS that the problems outlined above can be resolved during construction and that the Dresden 2 reactor can be built and operated at the proposed site without undue risk to the health and safety of the public.

Sincerely yours,

ORIGINAL SIGNED BY
W. D. MANLY

W. D. Manly
Chairman

References attached.

References:

1. Letter dated April 15, 1965 from Murray Joslin, Commonwealth Edison Company, to Atomic Energy Commission, with attachment "Application for Construction Permit and Operating License for Dresden Unit 2".

2. Dresden Nuclear Power Station, Unit 2, Plant Design and Analysis Report, Volume III, Plant Site and Environs, Commonwealth Edison Company, undated, received April 19, 1965.

3. Letter dated May 17, 1965 from Murray Joslin, Commonwealth Edison Company, to Dr. R. L. Doan, AEC Division of Reactor Licensing, transmitting Dresden Nuclear Power Station, Unit 2, Plant Design and Analysis Report, Volume I and Volume II, Drawings.

4. Letter dated July 9, 1965 from Murray Joslin, Commonwealth Edison Company to Dr. R. L. Doan, AEC Division of Reactor Licensing, transmitting Amendment No. 1 to the Dresden Nuclear Power Station Unit 2 Plant Design and Analysis Report, dated July 12, 1965.

5. Replacement page 2 of Amendment 1, undated, received July 20, 1965.

6. "Contents of Amendment No. 2", dated August 17, 1965 with attached errata and addenda, and transmitting Dresden Nuclear Power Station, Unit 2, Plant Design and Analysis Report, Amendment No. 2, Answers to AEC Questions, Commonwealth Edison Company.

7. Letter dated August 19, 1965 from Murray Joslin, Commonwealth Edison Company, to Dr. R. L. Doan, AEC Division of Reactor Licensing, with attachments constituting Amendment No. 3.

8. Letter dated September 16, 1965 from Murray Joslin, Commonwealth Edison Company, to Dr. R. L. Doan, AEC Division of Reactor Licensing, submitting Dresden Nuclear Power Station, Unit 2, Plant Design and Analysis Report, Amendment No. 4, Answers to AEC Questions and Dresden Unit 2 Fuel Allocation.

9. Letter dated October 21, 1965 from Murray Joslin, Commonwealth Edison Company, to Dr. R. L. Doan, AEC Division of Reactor Licensing, submitting Dresden Nuclear Power Station, Unit 2, Plant Design and Analysis Report, Amendment No. 5, Answers to AEC Questions.

Appendix B

Comments on

Dresden Nuclear Power Station Unit 2
Volumes I, II, III

Prepared by

Environmental Meteorological Research Branch
Office of Meteorological Research
June 17, 1965

A comprehensive analysis of meteorological data pertinent to atmospheric diffusion has been made for the Dresden area using data from the nearby Argonne Laboratory meteorological tower installation. We agree that the Argonne Data are climatologically representative of the Dresden site. From the Argonne data which covers a 5-year period, the following pertinent stability statistics result:

	0000-2400 hours (entire day)	1900-0700 hours (night-time)
Inversion	46%	71%
Neutral (-0.4 to 0.0 °C/140 ft)	28%	26%
Unstable	26%	3%

Similarly with regard to wind speed:

	0000-2400 hours	1900-0700 hours
Wind, 0 to 3 mph (19 feet)	17%	33%
Wind, 0 to 3 mph (150 feet)	4%	6%

The inversion frequency of 46% for a 140-ft height interval compares with the annual low-level inversion frequency of 30-35% obtained by Hosler (Monthly Weather Review, vol. 89, Sept. 1961) for the Dresden area for a 500-ft interval. A lower frequency for a greater height interval is to be expected. The low wind speed frequency of 17% at the 19-ft level at Argonne compares to 13% at the 15-ft level at

Dresden for 1 year of data. It is important to note that wind speeds in the 0-3 mph category decrease markedly at the 150-ft level at Argonne. Consequently, at a stack height of 300 ft. one would expect a similar low frequency of wind speeds less than 3 mph.

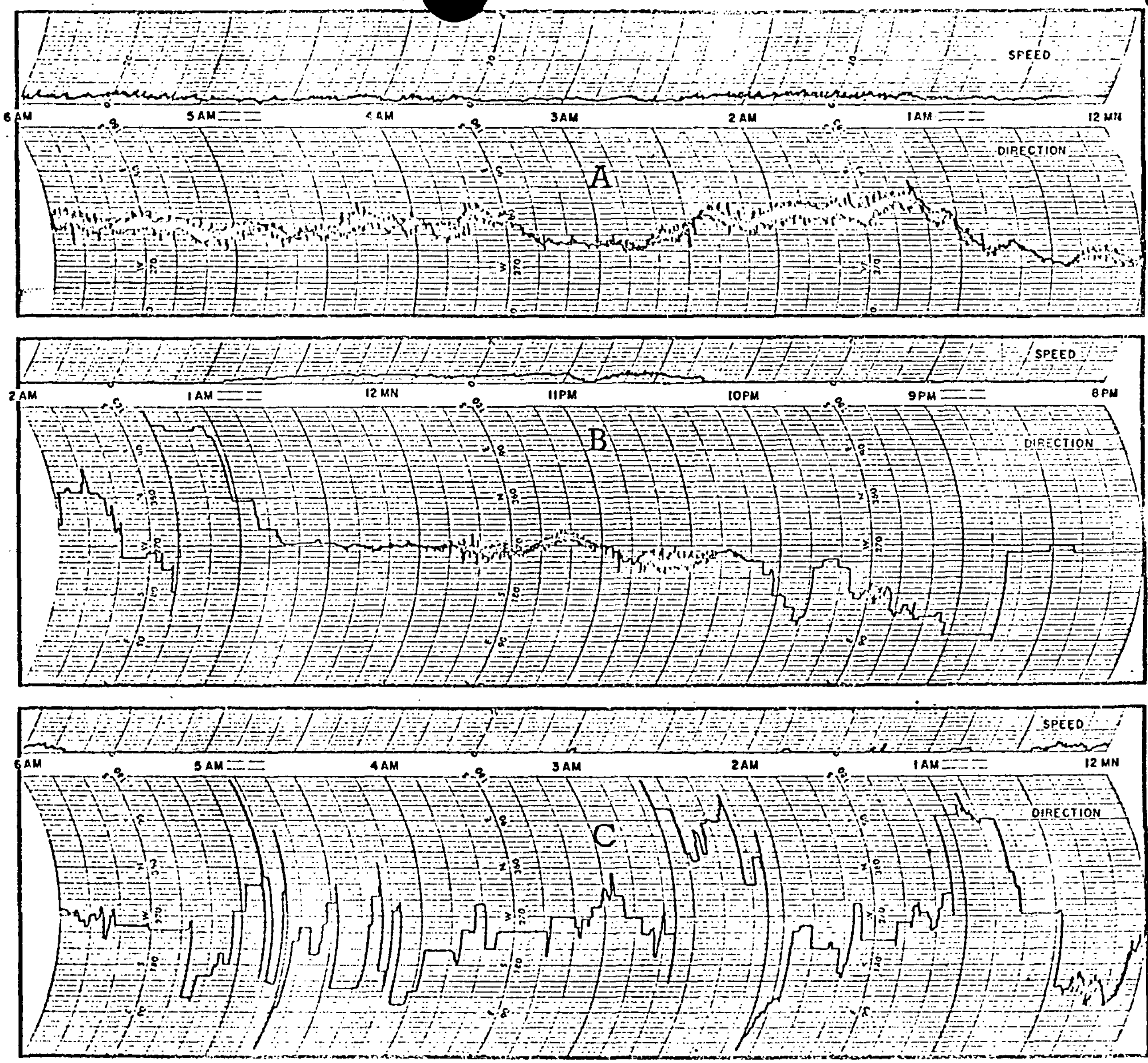
The site atmospheric diffusion characteristics as discussed in sections III-7 and XI-6 is largely based on an analysis of wind direction range data over an hour period from a one-year record of the Dresden "Aerovane" anemometer at a height of 15 feet. It should be noted that large direction fluctuations (range) are frequently associated with a decrease of wind speed to a value below the "starting speed threshold" of the vane of the anemometer. An instrument such as the Aerovane has a starting speed between 2 and 3 miles per hour. The attached figure, which is a sample Aerovane trace at a height of about 10 meters, illustrates the difficulty of using "range" data to estimate diffusion characteristics at very low wind speeds. Note that in trace A, with wind speeds above 3 mph, the wind direction range (difference in wind direction extremes) over an hour is about 80 degrees. In traces B and C, the vane is obviously stationary for long periods as indicated by the step-like changes in wind direction. If these extreme changes in direction under low wind speeds are included in the average range statistics, the application of these statistics to diffusion characteristics becomes less meaningful. Therefore, a graph such as Exhibit III-7-7 is probably not indicative of the horizontal spread, σ_y, of the cloud. Also, wind range statistics taken under inversion conditions at 15 feet are not necessarily applicable to such statistics at the 150-ft level of the Argonne tower or the assumed 600-ft effective stack height of the Dresden ventilation system. Regardless of height, if wind range statistics are to be used as a probability estimate of minimum diffusion rates, care should be taken not to average together the "steady" fluctuations in wind direction with the unsteady", step-like fluctuations characteristic of standard anemometers at low wind speeds.

A comparison of the horizontal and vertical cloud distributions (σ_y and σ_z) for the condition labeled VS-2 (very stable -2 mph) with those resulting from the Pasquill F condition used in TID 14844 shows the $\sigma_y - \sigma_z$ product at a distance of ½ mile to be about twice as great (therefore better diffusion) for the VS-2 condition, while at a distance of 5 miles the products are about equal. As a further comparison, using figure 6 of reference (1) cited on page XI-6-7, the σ_y versus time curve resulting from the TID-14844 assumption (C_y = .40, n = .50, u = 1 m/s) would be more nearly equal to a $\sigma_\theta \bar{u}$ value of 0.05 radian-meter/sec as opposed to the 0.16 value used in the VS-2 condition. As implied in the previous paragraph, the σ_θ statistics available in the report are probably not sufficiently meaningful in the low wind speed categories to be able to determine the probability of occurrence of this parameter.

The most critical parameter in the calculation of the downwind ground concentrations summarized in Table XI-7 is the assumption of an appreciable effective stack height. At first glance, it would appear that the very stable-2 mph condition is not the controlling case with regard to the off-site dose. In fact, with the use of a 170 m effective stack height and the very conservative value for vertical mixing in the VS-2 condition, it is safe to say that the cloud never reaches the ground within the first ten miles despite calculated dilution factors such as $(10)^{-120}$! However, using TID-14844 meteorological assumptions, an effective stack height of 170 meters and Sutton's continuous point source equation, a maximum ground concentration of 1.4×10^{-6} μc-sec/cc per curie at a distance of about 20 miles is computed. Thus, with these latter assumptions (TID-14844 with H = 170 m) which we feel are not unreasonably conservative and can be justified as well as the VS-2 assumptions, the ground concentration at 20 miles approach the point at which it becomes the controlling value.

In summary, it is felt that the Dresden site is typical of a continental, non-mountainous, non-desert location in the United States with a probability of having inversion conditions with low surface wind speeds about 20% of the time, during the night half of the day. The diffusion model used in the report is extremely sensitive especially for stable conditions, to the assumed effective stack height. Using the TID-14844 meteorological model and an effective stack height of 170 m, controlling concentrations can be found at distances of 20 miles, which is not apparent from either of the air concentration summaries found in Chapter XI. If for any reason, a ground release of effluent can be postulated, concentrations at a distance of ½ mile could well be on the order of 10^{-3} μc-sec/cc per curie released, which is three orders of magnitude greater than any values found in the report.

Attachment


SPEED
6 AM
5 AM
4 AM
3 AM
2 AM
1 AM
12 MN
DIRECTION
A
SPEED
2 AM
1 AM
12 MN
11PM
10PM
9PM
8 PM
DIRECTION
B
SPEED
6 AM
5 AM
4 AM
3 AM
2 AM
1 AM
12 MN
DIRECTION
C

Appendix C

IN REPLY REFER TO:
4030 0001

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON xx D.C. 20242

426

SEP 28 1965

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
4915 St. Elmo Avenue
Bethesda, Maryland

Dear Mr. Price:

Transmitted herewith are two copies of our review of the geology and hydrology of the Dresden site, Grundy County, Illinois.

The report was prepared in our Water Resources Division and was reviewed and approved in our Geologic Division. A draft of this report was sent to you with our letter of September 13; the changes requested on September 24 by Mr. Case and Mr. Hadlock have been made in this final version.

We have no objection to your making this report a part of the public record.

Sincerely yours,

Arthur A Baker

Acting Director

Enclosure

426

Rec'd Off. Dir. of Reg.
Date 9-29-65
Time 12:2
By

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON D.C. 20242

Geology and Hydrology of the Site of the Dresden No. 2 Unit,
a Proposed Nuclear Power Plant, Grundy County, Illinois
(AEC Docket No. 50-237)

By

E. L. Meyer and Alfred Clebsch, Jr.

Introduction

This statement is based on a review of information supplied in the Plant Design and Analysis Report for the Dresden Nuclear Power Station, Unit 2, compiled by Commonwealth Edison Company, and supplemented by further checking of pertinent published geologic and hydrologic reports. The exploratory drilling, testing, and analysis of geologic data by Dames and Moore and the Illinois State Geological Survey appears accurate and thorough. Field inspection of the site was not considered necessary.

Geology

Earth materials of concern at the site include thin unconsolidated glacial deposits of Pleistocene age and sedimentary rocks of Pennsylvanian and Ordovician ages.

Discontinuous patches of sandstone of the Pottsville(?) Formation of Pennsylvanian age underlie the glacial deposits; in the area of the No. 2 unit the sandstone is 40 to 50 feet thick. Beneath the sandstone the Divine Limestone of Lamar and Willman, 1931 (18 to 35 feet in thickness) and Maquoketa Shale of Ordovician age extend to maximum depths of 110 to 130 feet below the surface. The contact between the Divine and the Maquoketa ranges from 430 to 475 feet above sea level. Thus the plant foundation at about 470 feet above sea level will rest on Divine Limestone and the underlying Maquoketa Shale. Although some of the test specimens of shale have low values of ultimate compressive strength (Pl. III - 2 - 15) the rock in general seems to be adequate as a foundation for heavy structures. If major zones of weak rock are encountered in the shale during excavation, these rocks may have to be replaced with materials having higher ultimate compressive strength. Such conditions would not affect the suitability of the site.

Faults have been found in the Maquoketa Shale in cores from three holes. In other borings departures from the assumed regional southeasterly dip of the Maquoketa Shale suggest the presence of other faults that cut the Maquoketa Shale and probably extend into the overlying Divine Limestone. The Pottsville sandstone at the site is not faulted.

In the absence of faults in the Pottsville sandstone that can be related to faults in the Maquoketa Shale at the site it is inferred that the latter faults occurred subsequent to the deposition of the Divine Limestone, but prior to the deposition of the Pottsville sandstone, and therefore that faults through the site have been inactive since before Pennsylvanian time.

The Sandwich fault or fault zone strikes approximately S 60° E and passes about six miles northeast of the Dresden site; no information is available on its dip. The principal movement took place after Silurian time, and several authors have inferred that most of the movement took place after Mississippian time and before Pennsylvanian time (on the order of 300 million years ago). Because the youngest consolidated rocks in the area are of Pennsylvanian age, the history of the fault movement since the Pennsylvanian cannot be determined. Minor faults and folds in the Pennsylvanian rocks in the general area as well as gentle warping (Culver, 1923, p. 166 and 167) suggest some post-Pennsylvanian tectonic activity, the time of which cannot be ascertained. The faulting has also been explained as a result of differential compaction of underlying sedimentary rocks. There appear to be no reports of the displacement of deposits of Pleistocene age, and no ground displacement was reported after the earthquakes of 1909 and 1912, which were the most severe earthquakes experienced at the site in historical times. Therefore the probability of faulting through the site in the next 50 years is considered to be extremely remote. The relatively flat topography and firm foundation rocks preclude the occurrence of landslides.

Hydrology

The site is about 2,000 feet shoreward from the Dresden Island Dam pool in the Illinois River, immediately below the confluence of the Des Plaines and Kankakee Rivers.

Normal pool elevation at the site is 505 feet; the highest recorded stage for the river at the site since at least 1883 occurred in July, 1957, when the pool elevation reached 506.6 feet. A discharge of 93,700 cfs (cubic feet per second) was recorded for this flood at the Illinois River at Marseilles gage, 26 miles downstream. On the basis of flood frequency graphs for this gage shown by Mitchell (1954) a flood of that magnitude or greater has about a 25-year recurrence interval, or in other words, a 4 per cent chance of occurring in any one year.

Flow of the Illinois River consists of the natural runoff plus the diversions from Lake Michigan into the Chicago Sanitary and Ship Canal. During low flow the latter contributes a major part of the flow. The diversions from Lake Michigan into the canal are limited by order of the U. S. Supreme Court to an annual mean of 1,500 cfs (cubic feet per second) plus the total pumpage of the Chicago water supply system. The effect of the management of the diversion during the past several years has been to maintain a daily flow between 3,000 and 4,000 cfs in the canal whenever possible. This has the effect of stabilizing minimum flow in the Illinois River, shown in the report by the partial flow-duration series for the Illinois River at Marseilles, where flow is comparable to that at the site (Pls. III - 5 - 7 and III - 5 - 8). Natural low flow at the site can be characterized by the low flow pattern of the Kankakee River which contributes about 4/5 of the natural drainage area of the Illinois River at the site.

Records from the Kankakee River gage near Wilmington, about six miles upstream from the site, furnish an approximate picture of the natural low flow pattern at the site; in 27 years of record (1933-1960) minimum flow was 204 cfs, minimum daily flow, 319 cfs, and the lowest mean discharge for 30 consecutive days 376 cfs. Corresponding natural flows at the site could be estimated to be about 1/4 larger.

Public water supplies in towns in the vicinity of the site generally use ground water; surface water for public supplies in the general area is obtained from Lake Michigan and from the Kankakee River above the site, the Illinois River below the site, and the Vermillion River southwest of the site.

The nearest public water supply using the Illinois River is at Peoria, about 105 river miles downstream from the site. Peoria draws the major part of its supply from the Illinois River but also has a well field. The other nearby public supplies using surface water are at Kankakee (Kankakee River) about 28 miles southeast, Pontiac (Vermillion River) 35 miles southwest, Streator (Vermillion River) 40 miles west southwest, and Chicago (Lake Michigan) about 45 miles north northeast of the site.

Within a 15 mile radius of the site there are 11 public water supply systems, all using ground water. These supplies tap the St. Peter and Galesville Sandstones, at depths greater than 500 feet below the surface. A few of their wells also obtain a small fraction of their supply from the Galena Dolomite, of Ordovician age

The water resources currently in use in the area are quite well protected from accidental releases of radionuclides at the site. Contaminants released to the Illinois River are likely to take several days to reach Peoria at high flows and more than ten days at medium and low flows. The other surface water supplies are far enough from the site so that atmospheric dispersion would render insignificant the quantities of radionuclides estimated to escape in the major accidents as described in chapter XI, section 5. The public ground water supplies in the area are effectively separated from liquids deposited near the surface of the ground.

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IN REPLY REFER TO: File 800

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UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
WASHINGTON 25, D. C. 20240

SEP 8 1965

U.S. ATOMIC ENERGY COMM. REGULATORY SECTION 1965 SEP 10 AM 10 40 RECEIVED

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

In accordance with your request dated April 27, 1965, for our comments and recommendations on the application of the Commonwealth Edison Company for a license to construct and operate the Dresden Nuclear Power Station, Unit 2 (Docket No. 50-237), we are transmitting copies of a report by Dr. Theodore R. Rice and John P. Baptist entitled "A Preliminary Evaluation of Possible Effects on Fish and Shellfish of the Proposed Dresden Nuclear Power Station, Unit 2, Grundy County, Illinois."

We requested that Mr. William F. Carbine, Regional Director, Bureau of Commercial Fisheries, Ann Arbor, Michigan, discuss Dr. Rice's report and the application material transmitted by you, with local representatives of the Bureau of Sport Fisheries and Wildlife and the Illinois Department of Conservation, to obtain their comments in view of local knowledge.

On the basis of Dr. Rice's report and the comments received from other agencies, we believe that plans for control and disposal of radioactive wastes are adequate to protect fish and wildlife in the vicinity of the proposed plant. The recommendations on page 6 of Dr. Rice's report should be carried out by competent authorities to ensure that no adverse effects occur. We request that local Fish and Wildlife Service and Illinois Department of Conservation personnel be consulted in developing and reviewing the surveys needed to carry out these recommendations. Mr. Carbine, at the above address, would be happy to assist with the necessary coordination if desirable.

The Illinois Department of Conservation points out that it may be possible in the future to improve water quality in the Illinois River system. Operation of this and other plants along this river system should consider this possibility and nothing should be permitted which might make this task more difficult.

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We also request that the Illinois Department of Conservation be sent copies of all previous and future surveys conducted in connection with this plant and that they be kept informed of any future developments.

Fish and wildlife experts believe that thermal pollution may adversely affect aquatic resources. We recommend that appropriate studies of the effect of heated water be made and plant operation be modified to minimize any harmful thermal effect on aquatic life.

Although the Atomic Energy Commission feels that its regulatory authority over nuclear power plants includes only those hazards associated with radioactive materials, we urge that the hazards to fish and wildlife from thermal effects be called to the attention of the Commonwealth Edison Company and that they be encouraged to discuss this matter with representatives of the Illinois Department of Conservation and local Fish and Wildlife Service to develop measures to minimize this problem.

We are sending a copy of this letter and Dr. Rice's report to the Illinois Department of Conservation for their information.

Sincerely yours,

Clarence F. Pautzke

Clarence F. Pautzke
Commissioner

Enclosures

May 11, 1965

A PRELIMINARY EVALUATION OF POSSIBLE EFFECTS ON FISH AND SHELLFISH OF THE PROPOSED DRESDEN NUCLEAR POWER STATION, UNIT 2 GRUNDY COUNTY, ILLINOIS (DOCKET 50-237)

By

T. R. Rice, Director

and

J. P. Baptist, Fishery Biologist

Radiobiological Laboratory
Bureau of Commercial Fisheries
Beaufort, North Carolina

1. Introduction

The Commonwealth Edison Company of Chicago has applied to the Atomic Energy Commission for a construction permit and license to operate a nuclear power reactor in Grundy County, Illinois. The proposed reactor will be the second one erected at this site, which comprises 953 acres at the confluence of the Des Plaines and Kankakee Rivers. The first reactor (Dresden No. 1), a dual cycle boiling water type, has been in operation since 1960.

We understand that the jurisdiction of the AEC in the licensing and regulation of nuclear power reactors is limited to matters pertaining to radiological safety. For that reason, our comments in this report are divided into two categories. The first category pertains to radiological safety considerations, which are involved in the pending licensing proceeding. The second category contains our comments on the possible effects of increased water temperature on fishery organisms. Although these considerations are not within the jurisdiction of the AEC and are not involved in the pending AEC licensing proceeding, they may be of interest to appropriate state and local agencies and to the applicant.

The entry of radioactive materials into the aquatic environment, either by design or by accident, might conceivably result in adverse effects on the fisheries of the area. It was deemed advisable therefore that the Bureau of Commercial Fisheries of the U. S. Fish and Wildlife Service review the proposal and evaluate the possible effects of the operation of the proposed reactor on the fisheries. The present evaluation is based in part on information presented in "Plant Design and Analysis Report of Dresden Nuclear Power Station, Unit 2, vol. III" by the Commonwealth Edison Company, Chicago, Illinois.

2. Description of the Facility

Unit no. 2 will be a single cycle, forced circulation, boiling water reactor substantially similar, except for an increase in size and capability, to those authorized for construction at the Oyster Creek site in New Jersey and at Nine Mile Point in New York. The proposed reactor will operate initially at a power level of approximately 2,300 megawatts, thermal. Through the use of jet pumps, the recirculation coolant flow will require only a two-loop system. The reactor vessel and the recirculation system will be contained within a sealed steel pressure vessel, the drywell. The turbine, generator, feedwater heaters, condensate pumps and condensate mineralizer will be situated in the turbine building west of and adjacent to the turbine building for Dresden no. 1.

3. Radioactive Waste Treatment Facilities

The waste treatment facility will be located in the radwaste building next to the reactor building. The radwaste building will be a two-floor concrete structure containing the control, processing and storage areas necessary to operate the solid and liquid waste processing equipment.

Radioactive wastes will be gaseous, liquid and solid in form. Gases will be held up for decay of short-lived isotopes then routed to the existing Dresden stack for dispersion to the atmosphere. Liquid wastes will be collected, treated, stored and either returned to the condensate system, stored offsite or diluted with cooling water and discharged into the Illinois River. Wastes discharged into the river will not exceed maximum permissible concentrations listed in the Code of Federal Regulations, Title 10, part 20. Solid radioactive wastes will be shipped to an offsite disposal facility.

4. Hydrology

The Des Plaines and Kankakee Rivers comprise the upper part of the Illinois River system. The normal pool elevation due to the adjacent Dresden Island Lock and Dam is 505 feet. Nominal ground elevation at the site location of Unit 2 is about 516 feet. River flow data applicable to the Dresden site 1931-1964 show that river flow exceeded 3,000 cubic feet per second (CFS) or 98% of the days, and 6,000 cfs on 48% of the days.

The Des Plaines River below Lockport and the Illinois River are used for navigation, sewage disposal and dilution, and condenser cooling water for power plants. At and below Peoria, the Illinois River is also used for domestic water supply. The Kankakee River is not navigable and is used for domestic supply.

Detailed studies on chemical composition and biological conditions during 1961-1962 indicated that the lower river system was biologically degraded. The Dresden analysis states that the U. S. Public Health Service reported in 1963 that the upper portion of the river system shows a 9°C. net rise in temperature due to river usage.

5. <u>Fisheries of the Illinois River System</u>

Commercial fisheries of the Illinois River and its tributaries produced 2,208,600 pounds of fish during 1962, mostly carp, buffalofish, catfish, and crappie. (Power and Lyles, 1962). However, the contribution from the upper Illinois River to these catches was presumably small, judging from the reported difficulty in collecting fish samples in the vicinity of the Dresden site. Also, it has been reported by the Public Health Service that from July 1961 through July 1962 all stations from Wilmette, Chicago Harbor, and Calumet Harbor to a station 105 miles below the Dresden Dam were biologically degraded.

6. <u>Radioactivity Monitoring Programs</u>

Three agencies are involved in some phase of monitoring for the presence of radioactivity in the Dresden site environs. The Argonne National Laboratory has one monitoring point at Channahon, 3 miles north of the Dresden site. The State of Illinois Department of Health analyzes air and water from near the reactor site and from a point 9 miles downstream from the site. The Commonwealth Edison Company awarded a contract to Controls for Radiation, Inc. in 1962 to conduct a continuing monitoring program. A total of 3000 to 4000 radiochemical analyses are made yearly and include samples of air, water, slime, plankton, silt, vegetation, fish, and milk. Silt, slime, plankton, and fish samples are collected on a quarterly schedule and analyzed for gross alpha and gross beta radioactivity.

7. <u>Fate of Radionuclides in the Aquatic Environment</u>

When radionuclides are released into the aquatic environment various factors tend to dilute and disperse them while other factors tend to concentrate them. If the rate of dilution were the only consideration undoubtedly the maximum permissible concentrations of radionuclides which can be disposed of as wastes would be adequate criteria in determining the maximum safe rate of discharge. However, radioactive isotopes are adsorbed onto sediments and are concentrated by organisms which require many of the stable forms of these elements for their normal metabolic activities. In addition, some organisms concentrate radioisotopes not normally required but which are chemically similar to elements essential for metabolism. Furthermore, distribution of radionuclides can occur by their transmission from one organism to another through various trophic levels of the food web and by the migration of organisms from the area.

8. <u>Conclusions and Recommendations Concerning Radioactive Effluents</u>

The proposed reactor has been designed to operate with a minimum of environmental contamination by radioactive materials. However, these radioactive liquids must be released at a rate which will not exceed the maximum permissible limits set forth in title 10, part 20 of the Code of Federal Regulations.

It is concluded that the proposed nuclear reactor can be operated without harmful effects to the fisheries provided that a radiological monitoring program remains in effect during reactor operation, and the findings of this program are used to govern the discharge of radioactive material.

Although it is well established that certain levels of radioactive wastes can be discharged into the aquatic environment without adverse effects on the fisheries, it is essential to determine whether such discharge adversely affects the organisms in each specific area.

To insure that adequate safeguards are followed which will protect the fisheries from harm, certain requirements must be met. Therefore, it is recommended:

a. That ecological surveys be conducted 2 years prior to plant operation and continued on a regular basis after the plant starts operation to determine the effects of reactor effluents on plant and animal communities.

b. That extensive radiological monitoring of the biota, water, and sediments of the proximal aquatic environment be continued on a regular basis.

c. That hydrologic studies, such as those already carried out in the vicinity of the plant, be continued during plant operation to determine the extent of any changes which may occur due to discharge of radioactive effluents.

d. That consideration be given to the combined effects of effluent discharge from all existing and planned reactors along the shores of the river.

e. That the Radiobiological Laboratory be placed on the distribution list to receive copies of the survey and monitoring reports for review to assist other organizations in determining whether or not unsafe levels of radioactivity have been found in the water, sediments, or biota.

POSSIBLE EFFECTS OF INCREASED WATER TEMPERATURE ON FISHERY ORGANISMS

Large volumes of heated water discharged into an aquatic environment from a nuclear steam generating plant could result in a significant increase in the temperature of the environment near the plant. The temperature rise may or may not be sufficient to cause mortality among the organisms present, but subtle biological changes could occur causing long-term changes in the fisheries.

The thermal requirements of a fishery organism cannot be stated with any degree of accuracy. By "thermal requirements" here is meant the temperature limits which will permit survival at a level which allows for continuity of the species. These limits are influenced by season, age, size and other factors so that the thermal requirements would be quite variable and difficult to ascertain. As a controlling factor, the thermal requirement of a particular species becomes a level which will permit sufficient difference between resting and active metabolism to provide for essential activities (Brett, 1960). The increased energy demand of resting metabolism during elevated temperatures may rob an organism of the agility needed to capture its food. It has been proposed that the upper limit of required temperature for any species of fish should not exceed that which would curtail activity below 3/4 of the optimum, i.e., 3/4 of the maximum difference between active and resting metabolism (Brett, 1960).

Although a temperature rise in the aquatic environment may result in a change in species composition, increases in total productivity near warm water outlets from conventional power plants have been observed. Therefore it will be necessary to follow carefully any changes in total productivity in order to properly evaluate the effects on fishery organisms from discharged heated water.

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Appendix E

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OFFICE OF THE DIRECTOR

U.S. DEPARTMENT OF COMMERCE
COAST AND GEODETIC SURVEY
WASHINGTON SCIENCE CENTER
ROCKVILLE, MD. 20852

ENVIRONMENTAL SCIENCE SERVICES ADMINISTRATION

IN REPLY REFER TO: 68

September 17, 1965

Mr. Harold L. Price
Director of Regulations
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

In accordance with your request, we are forwarding 10 copies of a report on the seismicity of the Grundy County, Illinois area. The Survey has reviewed and evaluated the seismicity information presented by the Commonwealth Edison Company of Chicago, in their application for a construction permit and operating license for Dresden Unit 2.

If we may be of further assistance to you please do not hesitate to contact us.

Sincerely yours,

James C. Tison, Jr.

James C. Tison, Jr.
Rear Admiral, USESSA
Director

Enclosure

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U.S. ATOMIC ENERGY COMMISSION
REGULATORY MAIL SECTION

Rec'd Off. Dir. of Reg.
Date 9/21/65
Time 4:00
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REPORT ON THE SEISMICITY OF THE GRUNDY COUNTY, MORRIS, ILLINOIS AREA

At the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismology Division of the Coast and Geodetic Survey has evaluated the seismicity of the area around Grundy County, Morris, Illinois (Dresden Site). We find the report on seismology given by the applicant in the Plant Design and Analysis Report, Volume III, Section IV, is substantially correct although we consider the applicant's estimate of the maximum Modified Mercalli intensity to be expected at the site (VII to VIII), given on page III-1-5, to be slightly high.

The seismic history of the region since 1800 indicates that ten earthquakes have been perceptible in this area. The closest and most effective of these occurred on January 2, 1912 in the immediate area, probably less than 30 miles away, and is rated at intensity VI for a maximum acceleration of .07g in the period range from 0.3 to 0.6 seconds.

The shock of July 18, 1909 occurred at a distance of 75 miles and approached the previously cited earthquake in intensity at the site. Other earthquakes of note are the New

Madrid, Missouri series of 1811 and 1812 and the Charleston, South Carolina shock of 1886, all of which were quite distant and would have produced only minor intensities (III or less) at the site.

In view of the above, the Survey believes that the ground accelerations of more than 0.1g in the period range of 0.3 to 0.6 seconds will not be encountered at the Dresden site during the lifetime of the proposed facility.

EARTHQUAKE HISTORY OF GRUNDY COUNTY, ILLINOIS AREA
(DRESDEN SITE)

Date	Location	Miles	Maximum Intensity MM	Intensity Site MM	Felt Area Sq. Miles
1804 Aug. 20	Fort Dearbom, Ill., Chicago	50			30,000
1811 Dec. 16 1812 Jan. 23 1812 Feb. 07	New Madrid, Mo.	325	XII	IV**	2,000,000
1886 Aug. 31	Charleston, S.C.	400	X	III**	2,000,000
1895 Oct. 31	Charleston, S.C.	310	VIII	III**	1,000,000
1909 May 26	Northern Illinois	75±	VII	VI*	500,000
1909 July 18	Central Illinois	125±	VII	III**	40,000
1912 Jan. 02	Between Morris and Aurora, Ill.	30-	VI	VI*	40,000
1925 Feb. 28	Quebec, Canada	1000	VIII	III*	2,000,000

*Estimated from intensity reports in the vicinity
**Estimate based on the maximum intensity-distance relationship

Appendix F

NATHAN M. NEWMARK

Consulting Engineering Services

111 Talbot Laboratory, Urbana, Illinois

Report to AEC Regulatory Staff

ADEQUACY OF THE STRUCTURAL CRITERIA FOR THE DRESDEN NUCLEAR POWER STATION UNIT 2

by

N. M. Newmark and W. J. Hall

September 1965

ADEQUACY OF THE STRUCTURAL CRITERIA FOR THE DRESDEN NUCLEAR POWER STATION UNIT 2

by

N. M. Newmark and W. J. Hall

INTRODUCTION

This report concerns the adequacy of the containment structures and components for the 715 MWe net Dresden Nuclear Power Station Unit 2, hereafter referred to as Dresden Unit 2, for which application for a construction permit and operating license has been made to the United States Atomic Energy Commission by the Commonwealth Edison Company. The facility is located along the Illinois, Des Plaines, and Kankakee Rivers, in Grundy County, Illinois, about 40 miles southwest of Chicago, Illinois. Dresden Unit 2 will be constructed adjacent to, and to the west of, Dresden Unit 1.

Specifically, this report is concerned with evaluation of the design criteria that determine the ability of the primary and secondary containment systems to withstand a design earthquake of 0.1g maximum transient ground acceleration simultaneously with the other loads forming the basis of the containment design. The facility also is to be designed to withstand a 0.2g design earthquake loading to the extent of preserving the ability to maintain the plant in a safe shutdown condition. In addition, the seismic design criteria for Class I internal equipment and piping are reviewed.

This report is based on information and criteria set forth in the Plant Design and Analysis Reports (PDAR), and supplements thereto, as listed at the end of this report. In addition, we have participated in discussions with the applicant, its consultants, and the AEC Regulatory Staff, in which many of the design criteria were discussed in detail.

DESCRIPTION OF THE FACILITY

Dresden Unit 2 is described in the PDAR as a 2,255 MWt (715 MWe net) single-cycle forced circulation boiling water reactor that produces steam for direct use in the steam turbine. The fuel consists of UO_2 pellets in sealed Zircaloy-2 rods, and water serves as the moderator and coolant. The reactor vessel is about 21 ft in diameter, 68 ft high, and is to be made of SA-302 Grade B steel with Type 304 stainless steel interior cladding. The reactor vessel and the recirculation system are contained inside the drywell of a pressure suppression containment system.

The primary containment system consists of the drywell, vent pipes, and a structure shaped like a torus containing a pool of water; the center of the torus lies slightly below the bottom of the drywell. The drywell is a steel pressure vessel with a lower spherical portion about 66 ft in diameter and an upper cylindrical portion about 37 ft in diameter; the drywell is about 113 ft high, and the shell and head are to be made of SA-212 or SA-201 steel plate manufactured to A-300 requirements. The drywell is enclosed in reinforced concrete for shielding purposes, and to provide additional resistance to deformation and buckling; above the transition between the spherical and cylindrical portions of the drywell, we understand that the shell is separated from the concrete by a gap of several inches and that the backup filling material has not as yet been finally selected. Shielding at the top of the drywell is provided by a removable, segmented, reinforced concrete shield plug. The drywell contains one double-door air lock and one bolt hatch for access, in addition to the drywell head. The primary containment system is described in detail in Section V-3 of PDAR Volume 1.

The reactor building encloses the primary containment system; this building together with the standby gas treatment system and the 300-ft stack provide the secondary containment barrier. The reactor building houses the refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor auxiliary or service equipment. The reactor building is founded on rock. The substructure consists of poured-in-place reinforced concrete exterior walls up to the refueling floor. Above this floor level the structure is steel framed with insulated metal siding installed with sealed joints. Interlocked double doors provide entrance to the building. The secondary containment system is described in detail in Section V-4 of PDAR Volume 1.

SOURCES OF STRESSES IN CONTAINMENT STRUCTURES AND TYPE 1 COMPONENTS

The primary containment system, which includes the drywell, vents, torus, and penetrations, is to be designed for the following conditions: pressure suppression chamber (torus) and drywell internal design pressure, +62 psig, -2 psig; initial suppression chamber temperature rise, 50°F. As noted on page V-6-1 of PDAR Volume I, the aseismic design of the primary containment system, which is classified as a Class I--Critical Structure, will be based on dynamic analyses using response spectrum curves corresponding to a 0.1g design earthquake. It is further stated that the design will be such that a safe shutdown can be made during a ground motion of 0.2g, or in other words an earthquake with twice the intensity of the 0.1g design earthquake.

The secondary containment system, consisting of several parts as described earlier, is considered a Class I--Critical Structure. The reactor building is to be designed to withstand an internal pressure of 7 in. of water (about 1/4 psi) without structural failure. The aseismic design of the structure will be made for forces (supposedly coincident with dead load, snow load, and other applicable operating loads) corresponding to a design earthquake of 0.1g maximum ground acceleration. The design is to be made such that safe shutdown can be achieved for an earthquake motion of twice this intensity. The building is founded on rock.

Amendment No. 4 of the PDAR notes that the 300-ft stack, which is part of the secondary containment, is to be considered also as a Class I structure. Accordingly, the stack, which currently exists as a part of Dresden Unit 1, must be capable of resisting the forces arising from the same design earthquake in conjunction with other applicable loads.

The critical piping and equipment falling within the classification of Class I--Critical Structure or Class I--Critical Equipment, as listed in Section V-6 of Volume 1 of the PDAR, are to be designed to withstand the same seismic forces as noted earlier for the primary and secondary containment structures, in conjunction with other applicable loadings.

COMMENTS ON ADEQUACY OF DESIGN

Seismic Design Criteria -- In connection with the selection of the design earthquake, we agree with the approach adopted, namely that of basic design for a design earthquake of 0.1g, with the provision that a safe shutdown can be made for an earthquake of twice this intensity. We are in agreement with the soundness of this approach as presented by the applicant.

The design spectrum presented in Fig. 54 of PDAR Volume 3, which is identical to Fig. II-15-1 of Amendment No. 2, was examined in detail by us and appears to be acceptable in the light of a comparison with the 1940 El Centro earthquake. The latter has a maximum acceleration of 0.33g; the design earthquake used is approximately 0.3 times the intensity of the El Centro earthquake.

In Amendment No. 4 to the PDAR, in reply to Question 19, it is noted that both horizontal and vertical earthquake loads are included in each of the three representative examples, and that the vertical acceleration is taken as 2/3 the horizontal ground acceleration. We interpret this statement to mean that the same relative values of vertical to horizontal earthquake excitations will be used in all cases where seismic design is applicable.

In adding stresses arising from the different types of design loadings, that is dead load, pressure, wind, earthquake excitation, thermal effects, etc., in designing the containment structures and associated equipment, we believe that it is necessary to add directly (in terms of absolute numerical values) the stresses due to horizontal earthquake motions to those due to vertical earthquake motions, and to those due to pressure, temperature, dead load, and other operating loads as may be appropriate. On page II-15-2 of Amendment No. 2 it is stated that "for the design of Class I structures and equipment, the maximum horizontal acceleration and the maximum vertical acceleration will be considered to occur simultaneously. The resulting seismic stresses for the two motions will be combined linearly." In a discussion with representatives of General Electric and John A. Blume and Associates, we have ascertained that the interpretation of the foregoing statement is that maximum stresses which occur simultaneously at a particular location will be added directly in arriving at, or checking, the design. We concur in the approach.

For the earthquake of 0.2g maximum acceleration, it is indicated in Amendment No. 2 that in cases in which the material or structural elements are stressed beyond the yield point, calculations will be made to ensure that the energy absorption capacity available is greater than that which would correspond to the energy input from the earthquake. This approach appears reasonable to us in terms of limiting the deflections or distortions to permit proper functioning of critical pieces of the structure or equipment that are vital to a safe shutdown.

Two tables of damping coefficients are listed in the reports, one on page V-6-2 of PDAR Volume 1, and another on page II-15-2 of Amendment No. 2. We have been advised by representatives of General Electric and John A. Blume and Associates that the table of values given on page V-6-2 of PDAR Volume 1 will be those used in the design. We concur that the values given there are reasonable and conservative for use in the present design.

The seismic criteria employed in the design of the existing 300-ft stack, now a part of Dresden Unit 1, are described in Amendments No. 2 and 4. The procedures employed in the early design are not as accurate as the more rigorous analysis procedures proposed by the applicant for Dresden Unit 2. Amendment No. 4 indicates that a rigorous seismic dynamic analysis has been made of the existing stack; although only a brief summary of the results is presented, it appears that the stack possibly is satisfactory in its present state, or could be made so with only minor modifications, in terms of meeting the Class I--Critical Structure design criteria. Interestingly, the base shear in the more rigorous analysis corresponded to 6.7 percent of the total weight of the stack, in contrast to 3.3 percent which was all that was required by the earlier procedure. The stack is noted to be relatively safe against overturning, and it is assumed, in accordance with presently accepted procedures, that the stresses arising from the overturning moments at various levels in the stack are combined directly with other applicable stresses in checking the design.

It is indicated that the earthquake design will be based on ordinary allowable stresses as set forth in the applicable codes, and that one-third increase in the allowable working stresses because of the earthquake loading will not be used. Furthermore, it is indicated that Class II items will be designed following the normal practice for the design of power plants; as a minimum the seismic design will not be less than that given in the "Uniform Building Code" for Zone 1. We concur in these approaches.

No details are given concerning the possible strengthening of the areas around the penetrations of the containment, particularly the primary containment. Especially in the case of the large penetrations, care should be taken to ensure that these items will maintain the required strength and ductility under earthquake loading.

Primary and Secondary Containment Structures - General criteria covering the design of the primary and secondary containment structures are covered in various sections of the PDAR and supplements thereto. In Table 19-1 of Amendment No. 4 the allowable stresses for two combinations of loadings for the drywell (part of the primary containment system) are listed. The stresses noted appear to be in accordance with the applicable ASME codes, wherein rather high pseudo-elastic computed stress values are allowed for some stress combinations at locations of structural discontinuity, in the realization that local yielding will take place.

The allowable stresses for the reactor building for several load combinations are presented in Table 19-2 of Amendment No. 4. These appear to be in agreement with applicable codes, or in all other cases appear reasonable to us.

Class I Piping - Throughout the PDAR, frequent reference is made to meeting ASME and ASA code provisions. A tabulation of allowable stresses for a typical Class I piping situation is presented in Table 19-3 of Amendment No. 4. The stress values presented are in accordance with the ASA B31.1-1955 Code for Pressure Piping, and also permit rather high stresses for self-limiting stresses such as thermal stresses. In general the approach and values given appear reasonable.

The details of the pipe penetration design are discussed on page V-3-3 of PDAR Volume 1 and Fig. 48 of PDAR Volume 3, and appear to be reasonable and consistent with other designs of this type for both high-temperature and cold lines.

It is assumed that the critical Type 1 piping tie-downs and supports will be adequate to resist the appropriate design loadings, including seismic loadings, and any jet thrusts arising from possible broken pipes.

CONCLUSIONS

The design goal is to provide serviceable structures and components with a reserve of strength and ductility (a margin of safety) that will permit the structures and components to behave successfully under possible extreme loadings. On the basis of the information with which we have been supplied, we believe the design criteria outlined for the primary containment, secondary containment structures, and Type 1 piping, will provide an adequate margin of safety for seismic resistance.

REFERENCES

1. "Plant Design and Analysis Report -- Volume 1, "Dresden Nuclear Power Station Unit 2, Commonwealth Edison Company, 1965.

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3. "Plant Design and Analysis Report -- Volume 3 -- Plant Site and Environs," Dresden Nuclear Power Station Unit 2, Commonwealth Edison Company, 1965.

4. "Plant Design and Analysis Report -- Amendment No. 2 -- Answers to AEC Questions," Dresden Nuclear Power Station Unit 2, Commonwealth Edison Company, 1965.

5. "Plant Design and Analysis Report -- Amendment No. 4 -- Answers to AEC Questions," Dresden Nuclear Power Station Unit 2, Commonwealth Edison Company, 1965.