October 17, 1969

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-237

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1.0 INTRODUCTION

The Commonwealth Edison Company (applicant) submitted a Safety Analysis Report, dated November 17, 1967, as an amendment to its applications requesting provisional operating licenses for Units 2 and 3 of the Dresden Nuclear Power Station (facilities). These facilities, each of which will utilize a single-cycle, forced circulation General Electric boiling water reactor (BWR), have been under construction since the Commission issued a construction permit for Unit 2 on January 10, 1966, and a construction permit for Unit 3 on October 14, 1966. The facilities are located on a 953-acre site in the northeast quarter of the Goose Lake Township, Grundy County, Illinois. The site is adjacent to the Illinois River at the point where it is formed by the confluence of the Des Plaines and Kankakee Rivers. Located on the same site is Dresden Unit 1 (Docket No. 50-10) which has been in operation since issuance of an operating license by the Commission in 1959. Under construction at a separate, adjacent site is the General Electric Company's Midwest Fuel Recovery Plant (Docket 50-268).

Our technical safety review of the design of the virtually identical facilities was conducted concurrently and has been based on the Safety Analysis Report and Amendments 7 through 20 of the application for Unit 2 and 8 through 21 of the application for licenses for Unit 3. All of these documents are available for review at the Atomic Energy Commission's Public Document Room, 1717 H Street, Washington, D. C. In the course of the review, we have held numerous meetings with the applicant to discuss and clarify the technical material submitted. In addition to our review, the Advisory Committee on Reactor Safeguards (ACRS) reviewed the application as it applies to Unit 2 and met with both the applicant and us to discuss the facility. The ACRS report on Unit 2, dated September 10, 1969, is attached to this safety evaluation. The ACRS will review the application as it applies to Unit 3 when Unit 3 construction is nearing completion.

Our evaluation of overall performance of Unit 2 was based on a thermal power level of 2527 Mw, which will be the licensed power level. Our evaluation for the construction permit was based on a thermal power level of 2255 Mw, although the capability of the engineered safety features was reviewed for a power level of 2527 Mwt. Based on our evaluation of Unit 2, as presented in subsequent sections, we have concluded that Dresden Nuclear Power Station Unit 2 can be operated as proposed at power levels up to 2527 Mwt without endangering the health and safety of the public.

2.0 SITE AND ENVIRONMENT

2.1 Site Description

The Dresden Nuclear Power Station Unit 2 (facility or plant) site consists of approximately 953 acres located along the Illinois River about eight miles east of the City of Morris, the county seat of Grundy County. The site and adjacent areas are relatively flat except for a bluff across the river which is approximately 100 feet high. The minimum distance from the facility to the site boundary is 2200 feet. The distance to the nearest offsite residence is approximately 4000 feet. The Town of Channahan with a population of approximately 1200 is located 3.5 miles to the northeast, and the nearest population center is Joliet, centered 14 miles northeast of the site with a population of approximately 75,000. The applicant has proposed, and we agree, that a low population zone distance of five miles is acceptable for this site. The total population within this distance is approximately 3400, including summer visitors.

2.2 Meteorology

Meteorological data were obtained during 1968 and 1969 to supplement data taken earlier at the site and at the Argonne National Laboratory, 27 miles away. These data have been analyzed to obtain frequency distributions of stability, wind direction, and wind speed as functions of time of day and month of the year. The frequency of calms and inversions is typical for continental locations. The data justify the conservatism of the meteorological assumptions used in the accident analyses described in Section 6.0, and were used to derive limits for routine gaseous releases. Unit 2 is designed to withstand the effects of wind loadings and potential missiles resulting from tornadoes.

2.3 Hydrology

The site elevation is 516 feet as compared with the maximum historical flood elevation of 506.4 feet and the normal pool elevation of the river as controlled by the Dresden Dam of 505 feet. The facility is designed so that sufficient water to assure safe shutdown will be impounded in the intake and discharge canals for cooling in the event of a failure of the Dresden Dam and a subsequent lowering of the pool elevation of the rivers.

In the Technical Specifications, releases of radioactive effluents to the discharge canal are limited such that concentrations at the point of discharge do not exceed 10 CFR Part 20 limits for unrestricted areas. The closest downstream use of the Illinois River as a source of potable water is at Peoria, a distance of 100 river miles. We have concluded that the hydrological aspects of the site are acceptable.

2.4 Geology and Seismology

Additional study of the geological and seismological characteristics of the site since the construction permit for Unit 2 was issued has confirmed the original conclusions regarding the acceptability of the site. All major structures are founded on sound bedrock. The general nature of the rock at all depths is sound, with no evidence of faults or connected joints. We conclude that the geological and seismological aspects of the site are acceptable.

The facility was designed to withstand the effects of an earthquake corresponding to a maximum horizontal ground acceleration of 0.10g. Facility components and structures are designed such that the loads caused by an earthquake of this magnitude in combination with operating loads do not exceed code allowable stresses and for ground accelerations of 0.20g, there will be no loss of function of critical structures and components necessary to assure a safe and orderly shutdown. We and our consultant, Dr. Newmark (Nathan M. Newmark Consulting Engineering Services), have reviewed the seismic design of the facility.

As a result of our review, and as noted in the ACRS report, the applicant has agreed to supplement the analysis of the response of certain Class I structural and mechanical components. The applicant states that the results of these analyses will be in conformance with the design criteria or that any modifications that may be needed will be completed prior to fuel loading. On this basis we and our consultant conclude that the aseismic features of the facility are adequate.

We have required, and the applicant has agreed, that a strong-motion seismograph be installed to record data related to ground motion during a seismic event at its site. These data would be employed in the subsequent evaluation of the effects of the seismic event on the safe operation of the facility.

2.5 Environmental Radiation Monitoring

The requirements for the applicant's environmental radiation monitoring program are listed in the Technical Specifications. This program will include the monitoring of airborne particulates, gamma background, fallout, surface water, well water, bottom sediments, soil samples, and biological specimens. Recommendations from our consultant, the Fish and Wildlife Service of the U. S. Department of the Interior, have been incorporated into the applicant's environmental radiation monitoring program. We conclude that the applicant's program will be adequate for monitoring the radiological aspects of plant operation on the environs and assessing the health and safety aspects of the release of radioactivity to the environment from the operation of the plant.

3.0 FACILITY DESIGN

The following sections briefly describe the design of those features and systems of Units 2 and 3 that are important to safe operation. The two single-cycle, forced circulation boiling water reactors are located within a common reactor building which adjoins a common turbine building. The east half of the reactor building contains the Unit 2 reactor vessel, recirculation system, primary containment, emergency core cooling system, reactor auxiliary system, refueling equipment, and spent fuel storage facilities. Identical equipment for Unit 3 is located in the west half of the reactor building. A common new fuel storage vault is located in the Unit 2 section of the reactor building, and the two units share the same radioactive waste facilities. A common control room for Units 1, 2 and 3 is located at the juncture of the Unit 1 and Unit 2/3 turbine building. Since the applicant plans to operate Unit 2 prior to completion of construction of Unit 3, physical, electrical, and mechanical separation between the two units will be maintained to ensure that Unit 3 construction activities do not compromise the design bases for Unit 2 operation.

The principal design features and materials of construction of Units 2 and 3 are similar to those reviewed and approved for other boiling water reactors (e.g., the Oyster Creek facility, Docket No. 50-219, and the Nine Mile Point facility, Docket No. 50-220). Units 2 and 3 are the first BWR's to incorporate many of the features of the General Electric-designed BWR's now being proposed for construction. These features include higher power densities, the use of internal jet pumps, pressure vessels designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code, a steam-driven high pressure coolant injection system (HPCI) and a low pressure coolant injection system (LPCI) for core reflooding.

3.1 Reactor Design

3.1.1 General

The reactor core will contain 724 fuel assemblies, each of which consists of a 7 x 7 square array of cylindrical fuel rods enclosed within a Zircaloy-4 fuel channel. Fuel rods will consist of low-enrichment sintered UO_2 fuel pellets clad in Zircaloy-2 tubes. The outer diameter of each rod is 0.563 inch and its length is 144 inches. Fuel assemblies with similar configurations have been tested in operating BWR's. The reactor core will also contain 177 control rods of the bottom-entry type, moved vertically within the core by hydraulically operated drives. Control rods will consist of assemblies of 3/16-inch diameter stainless steel tubes filled with boron carbide (B_4C) powder and held in a cruciform array by a stainless steel sheath of 1/16-inch wall thickness. Similar, but shorter, rods are in use in Dresden Unit 1 and in other BWR's. Control curtains of boron stainless steel will be fixed between fuel channels during initial operation to supplement the reactivity worth of the control rods.

3.1.2 Core Thermal and Hydraulic Design

When the construction permits for Units 2 and 3 were issued, the core was rated for a thermal power level of 2255 Mwt, based on the critical heat flux limits contained in APED-3892, "Burnout Limit Curves for Boiling Water Reactors," issued in 1962. The applicant has now proposed operation at thermal power levels up to 2527 Mw (809 Mwe) based on the test data reported in APED-5286, "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," issued in 1966. We have concluded that the test conditions reported in APED-5286 were an adequate representation of expected operating conditions in Units 2 and 3, and that the limit lines established are acceptable for steady state conditions and anticipated transients. During normal operation, the minimum critical heat flux ratio (MCHFR) will be maintained in excess of 1.9.

As a result of the new heat flux correlation, the fuel operating conditions for Units 2 and 3 reflect linear heat generation rates and fuel exposures higher than those previously experienced by production fuel in BWR's. The peak linear heat generation rate during normal operation in Units 2 and 3 will be 17.5 kW/ft and the peak fuel exposure about 45,000 MWD/T. Because of the lack of irradiation data on full length fuel rods at the combination of fuel powers and exposures expected in Units 2 and 3 after approximately 15,000 MWD/T, the applicant will conduct a surveillance program on BWR fuel which operates beyond current experience with production fuel.

We have reviewed the applicant's analyses of the various transients that can be expected to occur during the operating lifetime of the plant. For all of the anticipated transients, the MCHFR remains above unity, which is assumed to be the threshold for fuel damage. The limiting transient was found to result from the instantaneous seizure of a recirculation pump during full power operation. For this case, the mismatch between heat flux and flow results in an MCHFR of about 1.05. Inadvertent continuous withdrawal of a single control rod until terminated by the rod block monitor results in an MCHFR of approximately 1.1. For this transient, the resulting heat generation rates are increased to approximately 21 kW/ft.

On the basis of our review, we conclude that adequate margin against fuel rod cladding damage is available for Unit 2.

3.1.3 Reactivity Control

Reactor power can be controlled by either movement of control rods or variation in reactor coolant recirculation system flow rate. A standby liquid control system is also provided as a backup shutdown system.

Control rods are used to bring the reactor through the full range of power (from shutdown to full power operation), to shape the reactor power distribution, and to compensate for changes in reactivity resulting from fuel burnup. Each control rod drive has separate control and scram devices. A common hydraulic pressure source for normal operation and a common dump volume for scram operation are used for the drives. On the basis of our review of the drive system design and the supporting evidence accumulated from operation of similar systems in Unit 1 and other reactors, we conclude that the installed system will meet the functional performance requirements for Unit 2 in a safe manner.

During operation at power levels below 10% of rated power, control rod worths are limited by the rod worth minimizer (RWM), a device which utilizes a computer to restrict control rod patterns such that rods which can be moved are worth no more than 1% Δ k, and the worth of no single control rod will exceed 2-1/2% Δ k. For reactor power levels in excess of 10% of rated power, the maximum control rod worth that could be established is 3.8% Δ k. Calculations of the consequences of a control-rod-drop accident (where a control rod equipped with a velocity limiter is assumed to fall by gravity from the core region with a rod worth of 3.8% Δk and reactor power in excess of 10% of rated) indicate that the peak fuel enthalpy is less than 200 cal/gm which is less than the enthalpy required for incipient fuel melting. Accordingly, we have concluded that use of the RWM is not required at power levels above 10%. A control-rod-ejection accident is precluded by the control rod housing support structure located below the reactor pressure vessel and similar to that installed in the Oyster Creek and Nine Mile Point reactors.

Reactor power can be controlled either manually or automatically through changes in the primary coolant recirculation flow rate. A load following capability is provided by the automatic load dispatch system described in Section 3.6.2. Analyses have shown that the most limiting thermal conditions occur at 100% of rated flow, assuming steady-state operation along the design flow control line.

The standby liquid control system is designed to bring the reactor to a cold shutdown condition from the full power steady-state operating condition at any time in core life independent of the control rod system capabilities. This requires that the liquid system provide about 10% Δk of shutdown reactivity. The liquid control system is designed to inject sufficient sodium pentaborate to provide 16% Δk of negative reactivity. The injection rate of the system is adequate to compensate for the effects of xenon burnup.

Based on the foregoing, we have concluded that the means provided for reactor control are adequate to compensate for changes in reactivity during operation and anticipated transients.

3.2 Primary Coolant System

The primary coolant system includes the reactor pressure vessel, recirculation loops, relief valves, safety valves and a single emergency isolation condenser. An in-service inspection program for the primary coolant system is described in the Technical Specifications. As noted in the ACRS letter, the applicant will review the program with us after five years of reactor operation, and modify it as necessary based on experience gained during operation. At that time, we will require the applicant to perform such inspections of components outside the reactor coolant pressure boundary as deemed necessary to provide continuing assurance of structural integrity. We conclude that the in-service inspection program, with provision for continuing review, is acceptable for this plant.

3.2.1 Reactor Pressure Vessel

The reactor pressure vessels of Units 2 and 3 are designed and fabricated for a pressure of 1250 psig and a temperature of 575°F in accordance with Section III, Class A of the ASME Boiler and Pressure Vessel Code and Code Case 1355, and with Code Case 1396 for the Unit 2 vessel. The inside diameter of the vessel is approximately 21 feet and the inside height is approximately 68 feet. The stress analysis required by the ASME Section III pressure vessel code was performed by the Babcock and Wilcox Company, the vessel fabricator, and was reviewed by General Electric. The vessels differ from previous vessels used for BWR's in that the core support structures are designed to accommodate jet pumps and the control rod drive housing stub tubes are made of Inconel. Also, a relatively new metal-joining method, the electroslag welding process, was used by the Babcock and Wilcox Company for the vertical welds in each shell course of the vessels. We and our consultants, P. Patriarca and E. C. Miller of the Oak Ridge National Laboratory, have evaluated the process and the properties of the vessel welds produced and conclude that they are acceptable for use in the Units 2 and 3 reactor vessels.

We have also reviewed the loadings on the core internals which would result from accident and earthquake conditions and have determined that stresses are within the limits of the ASME Section III Code and that the resultant deflections are limited so as to assure continued operation of the control rods and to preserve a coolable core geometry. The core internals have been analyzed to determine the potential for flow-induced vibrations; and during the startup testing, vibration measurements will be performed on the control rod guide tubes, in-core guide tubes, fuel channels, core plate, shroud, separators, recirculation loops and jet pumps. On the basis of our review of the applicant's analyses and testing program, we conclude that the design of the core internals is adequate.

3.2.2 Recirculation System

The reactor coolant recirculation system consists of two external loops with motor-driven centrifugal pumps and 20 jet pumps located in the reactor pressure vessel. The number of large vessel penetrations, the length of large diameter piping, the vessel blowdown rate, and reactor internal differential pressures are reduced from those in BWR facilities which do not employ jet pumps. Also, a capability for reflooding the vessel following an accident is provided.

Twenty jet pumps, 18 feet 6 inches high, are located in two symmetric groups between the vessel wall and the core shroud. Each pair of pumps is supplied driving flow from a 10-inch diameter riser pipe. The 10 riser pipes have individual vessel penetrations and connect to one of the two external recirculation loop manifolds. The jet pumps, which have no moving parts, consist of a nozzle that creates a high velocity jet and entrains suction flow, a throat section, and a diffuser. The nozzle and the throat are removable.

We have reviewed the analytical and experimental data on various operating modes as described in APED-5460, "Design and Performance of General Electric Boiling Water Reactor Jet Pumps." The flow behavior characteristics of the jet pumps of Units 2 and 3 under steady state and transient conditions are similar to those of plants not using jet pumps. At rated reactor power, each pump is designed to produce a flow of 4.9×10^6 lbs/hr at a head of 67 feet. The design suction and driving flows are 8300 and 4600 gpm, respectively.

Preoperational tests will be performed to demonstrate pump operation and to check for leakage from the nozzle to riser and throat to diffuser joints. Subsequent startup tests will include a calibration of jet pump instrumentation and a determination of head-flow characteristics, coastdown flows and single loop and equalizer line effects.

The stresses in the jet pumps and supporting assemblies will not exceed the limits specified in ASME Section III Code limits during normal operating conditions and anticipated transients. We have also examined the stresses resulting from seismic and accident loads, with subsequent operation of the HPCI system. Vibration analyses will be confirmed through frequency and amplitude measurements to be made during pre-operational and startup testing.

We have concluded that the design, analyses, and testing to date provide reasonable assurance of satisfactory jet pump performance in Unit 2.

3.2.3 Safety and Relief Valves

In contrast with earlier General Electric BWR plants, the safety valve capacity sizing is in accordance with Article 9, Section III of the ASME Code rather than Section I. Two safety valves are flange mounted on each of the four main steam lines, upstream of the first isolation valve. Individual valve capacity is approximately 630,000 lbs/hr, or slightly more than 6% of rated steam flow. Set points range from 1210 to 1240 psig. The applicant has provided an analysis of pressure transients which would result from a turbine trip assuming a turbine trip scram, a flux scram, and a pressure scram. The results of the analysis show that the Code limit of 1375 psig is not exceeded.

A total of five electromatic relief valves are provided which discharge directly into the pressure suppression pool, and operate automatically on high vessel pressure at 1125 psig, or upon initiation of automatic depressurization.

We conclude that these systems, when supplemented by the reactor protection system, provide adequate protection against over-pressurization of the reactor coolant boundary.

3.3 Primary Containment

3.3.1 General

As at the Oyster Creek and Nine Mile Point facilities, the primary containment system consists of a drywell, a pressure suppression chamber, a connecting vent system between the drywell and the water in the pressure suppression chamber, isolation valves, containment cooling systems, and other service equipment.

The drywell has a "light bulb" configuration with a free air volume of 158,000 ft^3. The pressure suppression chamber is a torus with a free air volume of 117,000 ft^3 and a water volume of 112,000 ft^3. The vent system between the drywell and the chamber consists of 8 vent pipes which connect with a ring header in the suppression chamber. Uniformly spaced around the ring header are 96 downcomer pipes which have a 3-foot 5-inch submergence in the water in the suppression chamber. The drywell, suppression chamber, and vent system tubes are designed for a 62 psig internal pressure at a temperature of 281°F. The suppression chamber design pressure was established to simplify pneumatic testing of the containment system.

3.3.2 Containment Design Basis

The loss-of-coolant accident produces calculated drywell and suppression chamber peak blowdown pressures of 47 psig and 27 psig, respectively. These peak pressures are calculated based on the hypothetical instantaneous severance of a 28-inch recirculation line, with the equalizer line open between the recirculation loops. Under normal operating conditions the equalizer line would be closed, thereby reducing the equivalent break area from 5.6 to 4.1 ft^2. Based on analytical models and experimental data taken at the Moss Landing test facility, the primary containment system will have a significant margin above the peak blowdown pressures calculated for the recirculation line break.

The applicant's analytical methods are the same as those for Oyster Creek and Nine Mile Point and have been checked against the results of the Moss Landing tests. A comparison of calculated and measured drywell pressures shows that the model predicts pressures higher than most of the test data. For the vent-to-break area ratio of 51 for Units 2 and 3, the agreement is quite close. Because of this agreement and the large pressure margin in both the drywell and suppression chamber designs, we consider the primary containment system adequate.

3.3.3 Mechanical Design

The drywell is made of A212 Grade B, made to A300 requirements and was designed, fabricated and inspected in accordance with ASME Code Section III, Subsection B. The design load combinations for the drywell, including seismic loads and thrust and impingement loads, are acceptable. Analyses by the applicant have indicated that there are no missiles which can penetrate the containment liner, and the recirculation system has been provided with pipe restraints to protect the containment against the effects of pipe whip.

A system for detecting leakage from the primary coolant system has also been incorporated into the primary containment. The system consists of an air sampling system and an open drywell floor drain sump and closed equipment drain tank with associated pumps and piping. Leakage into these tanks is measured by monitoring the quantity of water which is automatically pumped from these tanks. We conclude that the applicant's leak detection system is adequate and will provide effective means to detect even small leakage from the primary system.

3.3.4 Primary Containment Leak Rate Testing

The containment design pressure is 62 psig. During construction testing each vessel was strength tested at 1.15 times design pressure. After this, the containment was leak tested to demonstrate that leakage would not exceed 0.1% per day at design pressure. After installation of all penetrations, integrated tests will be conducted at pressures in steps up to 48 psig to establish reference data for use in later surveillance testing at 25 psig.

The Technical Specifications require that the leakage rate at the peak accident pressure of 48 psig does not exceed 1.6 weight percent of the contained air per 24 hours. In addition, tests will be performed to assure that leakage from specified individual testable valves or penetrations does not exceed 5% of the allowable operational containment leakage limit and that leakage from any main steam isolation valve does not exceed 11.5 cfm at a test pressure of 25 psig. We conclude that the testing program is adequate to provide assurance of containment integrity throughout the service lifetime of the facility. As recommended by the ACRS in its report on the acceptability of Unit 2, the applicant has agreed to undertake a program to further reduce effects of leakage from the main steam isolation valves. We conclude that this action is adequate to assure the maintenance of a low leakage containment system.

3.3.5 Primary Containment Cooling System

The containment cooling system consists of two independent and redundant spray-cooling loops for post-accident containment heat removal. Each loop will pump water from the pressure suppression pool (torus) through individual heat exchangers (which are cooled by the service water system) into spray headers located in the containment drywell. The water spray from the headers removes heat from the drywell atmosphere, and flows by gravity back to the torus. The heat removal capacity for each heat exchanger is 102×10^6 Btu/hr at a river water temperature of 95°F which is adequate to prevent overheating of the torus water following a design basis accident. We conclude that the system is acceptable.

3.3.6 Containment Inerting System

The containment atmosphere control system is designed to maintain an inert atmosphere within the primary containment to preclude possible combustion of hydrogen that may be evolved by a metal-water reaction as a consequence of a highly unlikely loss-of-coolant accident. The containment is purged with nitrogen gas before reactor operation and the oxygen concentration is maintained at less than 5%. Maintaining the oxygen concentration at this value assures that flammable mixtures of hydrogen and oxygen will not occur as a result of a metal-water reaction.

The system is located external to the drywell. Piping and component design, up to and including the first two isolation valves, will meet the requirements for Class I structures. The system also will be used to detect gross leakage paths in the primary containment boundary. This assures a continuous monitoring of containment integrity during plant operation.

We will periodically review our requirement for inerting as operating experience and further knowledge from development work currently underway are obtained, and as other means of eliminating the hazards from accident-generated hydrogen are found. We conclude that the inerting system is acceptable.

3.3.7 Secondary Containment

The secondary containment, or reactor building, encloses the primary containment structure (drywell and absorption chamber). It consists of reinforced concrete substructures to the elevation of the refueling floor, topped by a conventional steel building frame with insulated metal siding.

The building contains the reactor servicing facilities, new and spent fuel storage facilities, and reactor auxiliary systems including the isolation condenser system, demineralizers, standby liquid control system, control rod hydraulic system, and the standby gas treatment system.

The standby gas treatment system is designed to minimize the release of radioactive materials to the environment during a loss-of-coolant accident or whenever a high level of radioactivity exists in the reactor building. The system consists of two low capacity exhaust fans and two filtering trains of gas and particulate filters. Each train is capable of limiting the leak rate to 100% of the reactor building volume per day under neutral wind conditions. The fans are sized to maintain the reactor building pressure at a negative pressure of 0.25 inch of water.

A test program will be conducted to demonstrate the design capability of the secondary containment. Additional secondary containment capability tests will be conducted during various meteorological conditions and at each refueling outage. The charcoal filters of the standby gas treatment system will be tested to demonstrate a halogen removal efficiency of not less than 99%, using freon gas. The particulate filters will be tested using DOP to demonstrate a particulate removal efficiency of not less than 99% for particulate matter larger than 0.3 micron.

We conclude that the design features and testing program for the reactor building and standby gas treatment system are acceptable.

Unit 3 construction activities will prevent completion of secondary containment above the common refueling floor of Units 2 and 3 until after the Unit 2 fuel loading. Therefore, the initial fuel loading and low power physics testing of Unit 2 will be accomplished without a completed secondary containment building. Secondary containment will be achieved in Unit 2 prior to operation above 5 Mwt by sealing all openings in the refueling floor, and in the wall below the floor and between the two units. The potential radiological consequences of a control-rod-drop accident in an unirradiated core and without secondary containment are described in Section 4.3. We have concluded that secondary containment is not required for the initial loading of Unit 2.

3.4 Emergency Core Cooling System (ECCS)

3.4.1 General

The ECCS subsystems that provide emergency core cooling capability in the unlikely event of a loss-of-coolant accident (LOCA) at Units 2 and 3 are the high pressure coolant injection (HPCI), auto relief, core spray, and low pressure coolant injection (LPCI) systems. These subsystems can be operated from either onsite or offsite electrical power systems. The normal feedwater system can also provide additional protection for loss-of-coolant accident, but only if offsite electrical power is available.

High pressure coolant injection capability is provided by the normal feedwater and the HPCI systems. These systems provide subcooled water to the reactor vessel which depressurizes the reactor vessel. The HPCI contains one high pressure pump which takes suction from either the suppression pool ring header or the condensate storage tank. The high pressure pump discharges through a reactor feedwater line and then through the feedwater sparger to the downcomer region of the vessel. The pump is driven by a steam turbine which is supplied with steam from the reactor vessel and which exhausts steam to the suppression pool.

In the event of a loss-of-coolant accident without high pressure coolant injection capability (i.e., the normal feedwater and HPCI are assumed to be unavailable), the auto-relief system causes the reactor vessel blowdown to occur in a time interval sufficiently short to permit core spray and/or LPCI operation before excessive clad heating occurs. The system consists of five electromatic pressure relief valves. These valves are located on the main steam lines inside the drywell. All five valves are programmed to operate on initiation of the auto-relief system. Four of the five valves are required for satisfactory performance.

The LPCI contains two piping loops with two low pressure pumps and one heat exchanger in each loop. The pumps take suction from the suppression pool ring header and discharge to the two recirculation loops of the reactor coolant system. For short-term coolant injection into the vessel, three of the four LPCI pumps must operate; for long-term coolant recirculation through a heat exchanger, two LPCI pumps can be used for heat removal capability via the containment spray cooling mode of operation. Pressure reduction can be achieved via the manually operated containment spray cooling mode of operation.

Each of the two core spray subsystems includes one full capacity low head pump which takes suction from the suppression pool ring header and one sparger which distributes the coolant above the top of the reactor core. The subsystems are independent of one another except for their common use of the ring header.

The pumping equipment for the various engineered safety features is located in the corner rooms of the reactor building. The applicant has provided seals and water-tight doors to the access areas in these rooms to preclude flooding in the event of excess water leakage. We conclude that this action increases the safety margin of the facility.

All piping, fittings, and supports are designed to the ASA B-31.1 Code. All welded joints in pipe and fabricated fittings are examined by radiography in accordance with Paragraph VW51, Section VIII of the ASME Code for pressure vessels. These actions are adequate to assure an acceptable level of quality.

3.4.2 Functional Performance

The ECCS design provides active component redundancy for short-term cooling (coolant injection) and two completely independent subsystems for long-term cooling (coolant recirculation). This assures that the failure of a single active component cannot prevent coolant injection, and that the failure of a single component, active or passive, cannot prevent coolant recirculation and replenishment. For small liquid breaks up to about 0.12 ft^2 in area, the HPCI can supply sufficient coolant to depressurize the vessel and cool the core, depending only on the LPCI for long-term recirculation. For liquid breaks between 0.12 and 0.2 ft^2, the depressurizing function of the HPCI and the coolant makeup function of either the LPCI or the core spray subsystem act in conjunction for effective core cooling. As a backup to the HPCI, five electromatic relief valves are provided to function as an automatic vessel depressurization system. This system is actuated by a coincidence of low-low reactor vessel water level and high drywell pressure.

For liquid breaks larger than about 0.2 ft^2 where no depressurization assistance is required, a core spray subsystem by itself or in conjunction with the LPCI can adequately terminate the cladding temperature transient. The LPCI subsystem is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is independent of the core spray subsystem; however, it does function in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI subsystem in combination with the core spray subsystem provides adequate cooling without assistance from the high pressure emergency core cooling subsystems for a range of break areas from approximately 0.2 ft^2 up to and including 5.62 ft^2, the latter corresponding to the double-ended break of the recirculation line.

The computer codes used for the analysis of the ECCS performance following an LOCA are quite similar to the codes we reviewed for the Oyster Creek and Nine Mile Point facilities. For the Dresden Unit 2 analyses, however, credit was taken for additional core cooling during the accident by coolant level swell and by heat transfer based on transient critical heat flux assumptions.

We have concluded that there is adequate confirmation for the blowdown calculations to justify use of the calculated water level. However, we have also concluded that because of the incorporation of a calculation of the transient critical heat flux which lacks experimental verification, the total revised LOCA analyses performed for Unit 2 have not retained adequate conservatism to balance known areas of uncertainty. We therefore have based our review of the performance of the emergency core cooling systems principally on the results of previous, more conservative analyses.

3.4.3 Conclusion

On the basis of our review, we concluded that the ECCS is acceptable because it will (a) limit the peak clad temperature to well below the clad melting temperature, (b) limit the fuel clad-water reaction to less than one percent of the total clad mass, (c) terminate the temperature transient before the core geometry necessary for core cooling is lost and before the clad is so embrittled as to fail upon quenching, and (d) reduce the core temperature and remove core decay heat for an extended period of time.

3.5 Protection and Emergency Electric Power Systems

3.5.1 General

Our review of the protection and emergency electric power systems of Units 2 and 3 encompassed the following: Reactor Trip System, Rod Block Monitor System, Refueling Interlock System, Engineered Safety Features Initiating Systems, Containment Isolation Initiating Systems, Reactor Protection and Engineered Safety Feature Installation Criteria, Radiation Monitoring System, Emergency Power System and Applicable Environmental Testing. The systems for Unit 2 are separate from and independent of like systems of Unit 3 except that both units share the standby gas treatment system and portions of the auxiliary electric power system.

The Commission's General Design Criteria and the Proposed IEEE Criteria for Nuclear Power Plant Protection Systems (IEEE-279, dated August 30, 1968) served, where applicable, as the bases for judging the adequacy of the protection and emergency electric power system. The design of the reactor trip system, refueling interlock system, and containment isolation initiating system is similar to that of the Oyster Creek and Nine Mile Point facilities, and we have concluded that the systems meet applicable criteria and are acceptable.

Studies are in progress on further means of preventing common mode failures from negating scram action, and on design features to make tolerable the consequences of failures to scram during anticipated transients. The applicant plans to incorporate such changes in the design of Dresden Units 2 and 3 as are appropriate based on consideration of the results of these studies.

3.5.2 Rod Block Monitor System

The Rod Block Monitor (RBM) system is designed to prevent local fuel damage in the event of improper control rod withdrawal starting from any permitted power and flow conditions. The RBM utilizes signals from in-core neutron detectors which are adjacent to each control rod and from the recirculation flow detectors. The system is effective during control rod selection and movement above 30% of rated power.

Our review identified single component failures which would preclude rod block action when required. Analyses by the applicant have shown that inadvertent control rod withdrawal would result in a critical heat flux ratio of less than 1.0 only for certain limiting control rod patterns. As a result, the Technical Specifications permit control rod withdrawal from such limiting control rod patterns only after operability of the RBM has been established. We conclude that such provisions reduce the likelihood of fuel rod failure in the event of an uncontrolled control rod withdrawal incident.

3.5.3 Instrumentation for Emergency Core Cooling Systems

3.5.3.1 Core Spray

The core spray system is initiated by high drywell pressure signals or low reactor water level signals in coincidence with low reactor pressure signals. The high drywell pressure and low reactor water level are each monitored by four instrument channels. Low reactor pressure is monitored by two instrument channels. These same initiating signals start the emergency diesel generators.

Each core spray loop is actuated and controlled by a separate logic matrix. The contacts of the high-drywell-pressure trip relay are arranged in a one-out-of-two-taken-twice logic, as are the contacts of the reactor low-water-level trip relay. The low-reactor-pressure relay contacts are arranged in a one-out-of-two logic. The low-reactor-pressure signal is also interlocked with the admission valve control circuitry for each core spray loop to prevent the opening of this valve until reactor pressure has been reduced sufficiently.

3.5.3.2 Low Pressure Coolant Injection System (LPCI)

The Low Pressure Coolant Injection (LPCI)/Containment Cooling System is initiated by the same signals and trip logic as described for the core spray. In addition, circuitry is provided to identify the condition of each reactor coolant recirculation loop and assure the selection of and the injection into the unbroken recirculation loop.

The containment spray system consists of the same components as the LPCI plus the additional valves and piping required to direct cooling water to the containment spray headers. These components are arranged in two loops. The admission valve for each loop is manually operated by a switch located in the control room. The remote manual controls for these valves are interlocked so that opening is not possible unless primary containment pressure is above 1 psig and reactor water level inside the core shroud is above 2/3 the core height.

3.5.3.3 High Pressure Coolant Injection System (HPCI)

The High Pressure Coolant Injection System (HPCI) is initiated by either low reactor water level or high containment pressure signals. These parameters are each monitored by four channels of instrumentation. Signals from each of the four instruments monitoring a parameter are arranged in a one-out-of-two-taken-twice logic. Further, turbine trip occurs on high turbine exhaust pressure, low pump suction pressure or high reactor water level.

3.5.3.4 Automatic Pressure Relief System (APRS)

The APRS is initiated by instrumentation which monitors high containment pressure and low reactor water level. Automatic blowdown requires both that a high drywell pressure and low-low water level signal persist for a two-minute period. In addition, the design prevents blowdown until the discharge pressure of at least one LPCI pump or one core spray pump exceeds 100 psig. This design provides direct assurance that the low pressure ECCS pumps are operating prior to automatic depressurization.

Four instrument channels monitor each initiating parameter. Two of the four channels monitoring each parameter are assigned to one-out-of-two logic matrices. The arrangement of these signals within each logic matrix is two-out-of-two (pressure and level) in coincidence with two-out-of-two (pressure and level). The trip in one of these coincidence signals is interlocked with, and permits the starting of, a timer which delays actuation of the relief valves to permit operator intervention and to allow the HPCI to restore water inventory. The operator can reset the timer before it times out. The timer action completes the initiation circuitry. Each trip logic matrix actuates all five electromatic relief valves. The applicant has agreed to modify the design prior to fuel loading so that no single failure will prevent manual actuation.

3.5.3.5 Conclusion

On the basis of our review, we conclude that the instrumentation for the Emergency Core Cooling System conforms to the criteria of IEEE-279 and to the AEC's proposed General Design Criteria and is acceptable.

3.5.4 Standby Gas Treatment System

The Standby Gas Treatment System is shared by Units 2 and 3, and is placed in operation by either (1) high radiation from the monitors located over the fuel pool or at the exhaust point of the reactor building ventilation system, (2) high containment drywell pressure, or (3) low reactor vessel water level. These signals initiate operation of one of the two subsystems through redundant logic circuits. Should the subsystem fail to operate, operation of the second subsystem would be initiated by other redundant logic circuits provided for this purpose.

Our review showed that the system satisfied the IEEE-279 criteria and applicable AEC General Design Criteria except with respect to the physical separation of redundant electrical components.

The applicant has agreed to provide physical separation of electrical components in accordance with IEEE-279 prior to operation above 5 Mwt. We conclude that with this modification, the system is acceptable.

3.5.5 Instrumentation Installation Criteria for Reactor Protection System and Engineered Safety Features

The design and installation of the protection systems comply with the requirements summarized below:

(1) Sensors are divided into four or more channels and their channel division is carried through to the protection system relay panels which consist of four separate panel sections having enclosures of steel.

(2) All protection systems wiring is run in rigid metallic conduits or solid trays with covers.

(3) Cables through drywell penetrations are so grouped that failure of all cables in a single penetration cannot disable any protective function.

(4) Routing of cables is such that damage to any single tray cannot disable any protective function.

(5) Sensors are arranged so that no single sensor or process sensing line failure in any mode can disable any protective function.

(6) Wiring to scram solenoids is grouped so that no failure within a single metallic enclosure can affect more than one of the four groups of control rods.

(7) Three-phase circuit breaker protection is provided.

Further, panels in the main control room are designed so that cables for core spray A and core spray B are in separate panels having steel end closures. Wiring for the LPCI system is similarly separated. The HPCI is also separated from the APRS in the same manner. Relay cabinets for these systems follow the same rules of separate cabinets for separate subsystems. Power supplies are separate for each core spray subsystem and the LPCI power supply is similarly separated, as are the HPCI and APRS systems. Interconnecting cables use separate trays or conduits so that no single wireway failure can disable the core cooling functions.

The basic design criteria for cable runs in trays or conduits include the following: precautionary measures for prevention of cable fires, containment of any fire to a confined area, and protection of cables against fire damage (other than electrically induced) in hazardous station areas.

We have concluded that the installation based on the criteria summarized above is satisfactory.

3.5.6 Process Radiation Monitoring

The radiation monitoring system installed in Units 2 and 3 meets the following criteria:

1. The operability of all monitors can be verified by means of test signals or radioactive test sources.

2. The systems satisfy the IEEE-279 criteria where automatic protective action is required. This redundancy is either inherent in the system or is provided by backup monitors.

3. All discharges of radioactive materials to the environment are continuously indicated and recorded in the control room.

4. All process and local area radiation monitors provide alarms or indications in the control room.

On the basis of our review, we have concluded that the radiation monitors are satisfactory.

3.5.7 Emergency Power System

Offsite power for Dresden 2 is obtained from the 138 KV switchyard through the reserve auxiliary transformer. Power is transmitted on the 345 KV system. Unit 3 both transmits and receives offsite power from the 345 KV system. Both units have a unit auxiliary transformer powered from the generator. The 138 KV and 345 KV switchyards are separated by approximately 1000 feet. Six transmission circuits emanate from the 138 KV switchyard and five transmission circuits from the 345 KV switchyard.

These transmission lines are routed to the distribution system via four separated rights-of-way. These transmission circuits (lines) are separated at their respective switchyards. This separation affords the means to isolate a bus section which is damaged by fire or mechanical fault without affecting the total switchyard. The applicant has conducted system stability studies and has concluded that the grid will be able to supply offsite power to the station in the unlikely event of the simultaneous loss of power from Units 2 and 3.

Within each Unit there are two independent buses. There is one diesel at each Unit specifically assigned to one bus at that Unit. The third diesel is shared by both Units and can energize either remaining bus automatically as required. In the event of an accident at Unit 2, high containment pressure signals or low reactor water level signals will initiate the start of the diesel generator for Unit 2 and the shared diesel generator. These signals prevent the shared diesel from being aligned to the non-accident unit. The diesel generators start and within 10 seconds reach rated voltage and frequency and are prepared to connect to their respective emergency buses. Loss of voltage at either emergency bus, detected by undervoltage relays, will open the respective supply breakers to offsite power sources and allow the respective diesel generator breaker to close. The diesel generators are then automatically loaded with the engineered safety feature in 30 seconds. Further, undervoltage on the emergency buses will initiate the start of the diesel generators and the shedding of all loads connected to these buses.

Each diesel generator is housed in a separate Class I area and has adequate capacity for emergency loads and/or shutdown loads for one unit. The diesel generators are each rated for continuous service at 2500 kW. The automatically energized loads total 1950 kW.

We conclude that the offsite and onsite power systems are acceptable.

3.5.8 Environmental Testing

The equipment within the primary containment that must function in an accident environment consists of a-c electric motor-operated valves (ECCS and Isolation valves) with their associated actuators and electrical cabling, the solenoid-actuated electromatic relief valves on the main steam lines, and certain instrumentation sensors. We have concluded that studies by the applicant provide reasonable assurance that equipment used in the reactor protection and engineered safety feature systems can perform their design functions in an accident environment.

3.6 Other Systems

3.6.1 Control Room

The control room contains all necessary controls and instrumentation for operation of the reactor, turbine-generator and auxiliary systems for the three units. The control room is designed to be occupied during design basis accident conditions as well as during normal operation. Although specific provisions were not made in the design, the equipment necessary to conduct safe shutdown of Units 2 and 3 can be operated remotely from outside the control room. During Unit 3 construction activities, a fireproof barrier will be maintained between the Unit 3 area of the control room and the area of Units 1 and 2.

The control room has adequate instrumentation and controls for controlling the reactor facilities in a safe manner. While all reactor protection and engineered safety features are automatic, facilities for manual operation of the safety features are also provided in the control room.

We have evaluated the design of the reactor control room with respect to the adequacy of the shielding for continuous occupancy in the event of the design basis accident, and with respect to the potential doses during ingress and egress subsequent to an accident. Our calculations show that adequate shielding has been provided to limit the whole body doses to an operator to less than 5 rem. We conclude that adequate control room shielding has been provided.

3.6.2 Load Control System

The load control system of Units 2 and 3 consists of a turbine control system and a recirculation flow control system. To obtain desired load following characteristics, a signal from the turbine governing system controls the reactor recirculation flow and directly accomplishes changes in reactor power. An automatic load dispatch system, whose function is to maintain adequate power generation for the applicant's grid, can be interconnected with the turbine control system at the discretion of the control room operator.

Our review of this system indicates that the reactor transients which would be expected during normal operation of the system and as a result of system failures would not result in fuel damage. As a result of our review, the applicant has added indicating lights at the reactor console to provide the operator with information to distinguish between a plant malfunction causing a load change and automatic dispatch system requirements. The use of the automatic dispatch system will be limited to the range from 70 to 100% of rated reactor power. The results of startup and power testing will be made available for our review prior to routine operation with the automatic load dispatch system. On this basis, we have concluded that the load control system is acceptable.

3.6.3 Radwaste Systems

The purpose of the radwaste system is to treat and dispose of all types of solids, liquid, and gaseous radioactive wastes accumulated during operation of the facility.

The solid radwaste system serves to collect, process, and package items such as filter sludge, spent resins, and equipment originating in the primary system for offsite disposal. The material is dewatered in a centrifuge, compressed into 55-gallon drums, or mixed with concrete in preparation for shipment, depending on the quantity and activity level.

The gaseous radioactive waste control system is designed to process non-condensible gaseous products from the main condenser to limit fission product release to the environment. A 30-minute holdup capability is provided to allow radioactive decay of short lived products prior to stack release. The stack gas is continually monitored. The Technical Specifications of Units 1, 2 and 3, and the Midwest Fuel Recovery Plant will assure that total offsite doses resulting from combined operation of all facilities are not in excess of the limits for radioactivity releases from a site given in 10 CFR Part 20.

The liquid radioactive waste system collects, treats, and disposes of all liquid radioactive wastes generated within the facility. All liquid wastes are collected, sampled and discharged on a batch basis and monitored so that inadvertent discharge of high activity waste is unlikely.

We conclude that these systems are adequate to assure that the 10 CFR Part 20 limits will not be exceeded.

3.6.4 Fuel Handling and Storage

Fuel handling operations are carried out using facilities provided for unloading and storing of new fuel in the reactor building, transferring and unloading of new assemblies into the reactor core, transfer of spent fuel assemblies from within the reactor vessel to storage in the spent fuel pool, and offsite shipment of spent fuel assemblies for reprocessing in a specially designed cask.

During refueling, transport to the spent fuel storage pool, and during storage, spent fuel will be continuously submerged in water. The spent fuel storage racks in the pit are arranged to assure a subcritical array. During refueling and storage, personnel will be protected by water and/or concrete shielding. Systems are provided to monitor spent fuel pool water temperatures and activity. Refueling platform travel switches are interlocked with control rod position indicators to assure that all control rods are inserted whenever fuel is being carried over the reactor core.

On the basis of our review, we have concluded that the provisions for fuel handling and storage are acceptable.

4.0 ANALYSES OF DESIGN BASIS ACCIDENTS

Four major postulated accident situations were considered as design basis accidents to assess the adequacy of the Unit 2 engineered safety features to control the possible escape of fission products from the facility. The design basis accidents analyzed were: (1) control-rod-drop, (2) refueling, (3) steam-line-break, and (4) loss-of-coolant accidents. In addition, we examined postulated accidents which could result from pipe or component ruptures within emergency core cooling subsystems such as the core spray, LPCI, and HPCI systems. Our evaluation of these accidents showed that effective core cooling would be maintained and that the resultant radiological consequences were significantly less than those calculated for the design basis accidents.

The results of our analyses for the design basis accidents are summarized in the following sections and the doses which we have calculated using conservative assumptions are summarized in Table 4.0. The doses resulting from these postulated accidents are well within 10 CFR Part 100 guideline values.

TABLE 4.0

CALCULATED DOSES IN THE EVENT OF POSTULATED ACCIDENTS AT UNIT 2 OR 3

Accident	Two Hour Dose at Site Boundary (rem)		30 Day Dose At The Low Population Zone (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss of Coolant	185	8	90	2
Refueling	25	<1	8	1
Control Rod Drop	55	1	1	<1
Steam Line Break (10 sec valve closure time)	25	<1	<1	<1

4.1 Loss of Coolant Inside the Drywell

In calculating the consequences of the loss-of-coolant accident, we have assumed fission product release fractions released from the core as suggested in Technical Information Document 14844, "Calculations of Distance Factors for Power and Test Reactor Sites," i.e., 100% of the noble gases, 50% of the halogens, and 1% of the solids. In addition, 50% of the halogens released from the core is assumed to plate out onto internal surfaces of the containment building or onto internal components. The primary containment was assumed to leak at a constant rate of 2.0 percent of the containment volume per day for the duration of the accident without consideration of the effects of decreasing pressure during the post-accident interval.

We have assumed a 90% halogen removal efficiency of the charcoal absorbers of the standby gas treatment system in the secondary containment building. In our analysis, we took the conservative approach of assuming that leakage from the drywell goes directly to the standby gas treatment system without mixing in the reactor building and then to the environs via the 310-foot stack.

Fumigation conditions were assumed for the first half hour exposure at the site boundary, followed by the most conservative unstable condition. The controlling location was found to be about 1300 meters northeast of the stack, at a 100-foot river bluff. In calculating the doses at the low population distance for the first 8 hours, we used a dilution factor, based on the curves for the various Pasquill types of meteorology for a 310-foot release height, that maximizes the calculated dose as a function of distance. For 8 to 24 hours this condition was assumed to continue, but the plume was spread uniformly in a 22-1/2 degree sector. For the next three days, the wind was assumed to continue blowing into the same sector, but diffusion conditions were varied so as to shift the location of the maximum concentration, and the wind speed was allowed to increase. After four days, similar diffusion conditions were used, but the wind was assumed to remain in the sector only 1/3 of the time.

In addition to the radiological consequences of an assumed loss-of-coolant accident, the potential consequences of radiolytic decomposition of water have been considered. Such decomposition would result in the production of gaseous hydrogen and oxygen in the containment atmosphere. If sufficient hydrogen and oxygen are produced by such a reaction, it is possible that a flammable mixture could be attained in the containment that if ignited would introduce an additional source of energy into the containment system. Preliminary studies by the applicant suggest that the extent of the decomposition reaction may be limited by back-reaction rates. This matter is undergoing thorough review by industry, Oak Ridge National Laboratory, Battelle Memorial Institute, and the Commission's Division of Reactor Licensing. We will evaluate further information as it becomes available and will require the applicant to take such action as deemed necessary to control the concentration of hydrogen in the containment.

4.2 Refueling Accident

In our evaluation of the refueling accident we assume that during the fuel handling operations, a fuel bundle falls with sufficient force to physically damage (perforate) 49 fuel rods (1 assembly) with consequent release of 20% of the noble gases and 10% of the halogens from the damaged rods into the reactor building. Ninety percent of the halogens released from the perforated fuel rods are assumed to remain in the refueling water. The remaining airborne fission products (20% of the noble gases and 1% of the halogens contained in the fuel) within the building are assumed to be discharged to the atmosphere through the standby gas treatment system (with an iodine filter removal efficiency of 90%) and through the stack over a two-hour period. It is assumed that the accident occurs 24 hours after shutdown. The meteorological conditions assumed are the same as described above for a loss-of-coolant accident.

Even in the extremely unlikely event that as many as nine fuel assemblies were to fail, as suggested in the applicant's analysis based on the conservation of energy following impact of a fuel assembly on the core, the doses would remain well below the 10 CFR Part 100 guideline values.

4.3 Control Rod Drop

In the control-rod-drop accident it is assumed that a bottom-entry rod has been fully inserted and has stuck in this position unknown to the reactor operator. It is then assumed that the drive becomes uncoupled and withdrawn from the rod. Subsequently, it is assumed that the rod falls out of the core inserting an amount of reactivity corresponding to the worth of the rod.

Hot standby is the worst operating condition at which the accident could happen both because a higher energy release is calculated for this condition and because a path for the unfiltered release of fission products could exist through the mechanical vacuum pump on the condenser. A rod reactivity worth of 2.5% Δk, the highest worth rod permitted by the Technical Specifications, was assumed in the analysis. This reactivity addition would result in a peak fuel energy density of about 220 cal/gm (average across the peak fuel pellet). Perforation of about 330 fuel rods is predicted.

We have evaluated the consequences of the control-rod-drop accident assuming that 330 fuel rods fail, releasing 100 percent of the noble gases and 50 percent of the halogens from the affected rods to the primary system. Of the halogens released from the affected rods, 90 percent are assumed to be retained in the primary system and one-half of the remaining halogens are assumed to be removed by plateout. All of the noble gases and 2.5% of the halogens would be released from the primary system through the condenser vacuum pump system to the atmosphere. A ground release was assumed with Type F conditions at 1 m/sec for the two-hour doses at the site boundary. At the low population distance, these conditions were assumed to continue for 8 hours, followed by spreading of the plume into a 22-1/2 degree sector from 8 to 24 hours. For this accident, the 24-hour time interval is the full course of the accident.

An automatic isolation valve has been installed on the discharge side of the condenser vacuum pump which would be closed by a high radiation signal from the steam line monitor to confine fission products released from the fuel to the primary system. The pump would also be tripped by these signals, thus providing a second barrier to the release of fission products. These features were considered in the calculations, and the resulting doses are well within the 10 CFR Part 100 guidelines.

The applicant has proposed, and we agree, to allow initial fuel loading without secondary containment. The rod-drop accident for the unirradiated core during the initial fuel loading of Unit 2 without secondary containment has been analyzed, assuming an energy generation of 82 MW-sec in the 200 fuel pins which perforate during the transient, the release fractions noted above and a factor of 2 for plateout in the building. The resulting offsite dose from a ground level release is approximately 2 rem thyroid, with a much smaller whole body dose. These doses are well within the 10 CFR Part 100 guidelines. Secondary containment integrity will be established prior to power operation of Unit 2.

4.4 Steam Line Break Outside Containment

The break of a main steam line outside of both the drywell and the reactor building represents a potential escape route for reactor coolant from the vessel to the atmosphere without passage through the reactor building and standby gas treatment system.

The steam line break would be sensed by either high steam flow or increased temperature in the pipe tunnel if the break occurred in this region. The steam line isolation valves would start to close within 0.5 second after the steam line break is sensed. The valves are designed for a closure time of from 3 to 10 seconds. In our analysis, we have assumed that valve closure time is increased to its maximum adjustment of 10 seconds. The meteorological considerations assumed for this accident are the same as for the control-rod-drop accident.

In order to assure that the doses that may result from a steam line break do not exceed 10 CFR Part 100 guidelines, it is necessary that no fuel rod perforations occur prior to closure of the main steam line isolation valves. Analyses have been provided that show fuel rod cladding perforations would be avoided for valve closure times, including instrument delay, as long as 10.5 seconds. In our opinion, these analyses appear reasonable and could support acceptance of valve closure times up to about 10 seconds. However, for additional margin to assure that fuel failure would not occur during the transient before the valves are closed, the Technical Specifications will require a valve closure time of not greater than 5 seconds. Further, the primary coolant total iodine fission product inventory is established at 20 μCi/cc, which corresponds to the allowed stack release rates. Using these assumptions, the two-hour thyroid dose would be reduced to approximately 10 rem.

4.5 Conclusion

On the basis of our evaluation, the radiological doses that could result from any of the design basis accidents are well within the guideline values given in 10 CFR Part 100.

5.0 EMERGENCY PLANNING

The applicant has described a comprehensive plan for coping with the consequences of an accident which might affect the general public. Arrangements to deal with radiological emergencies have been made with the responsible agencies of the state of Illinois and appropriate local officials.

Members of the applicant's onsite staff will furnish information to state and local officials concerning the release of fission products from the facilities. The applicant possesses the capability for providing offsite monitoring. They will also provide technical advice concerning the potential offsite effects throughout the course of any accident affecting the general public.

In addition, technical assistance is available through the Radiological Emergency Assistance Team program of the AEC. The applicant has established liaison with the team at the Chicago Operations Office of the AEC.

The applicant has made arrangements with two medical doctors trained in radiation medicine to provide medical consultant services. St. Josephs Hospital in Joliet has agreed to provide medical care for the Dresden Station and to make available such support as might be required in the event of an accident at the site, whether or not such an accident should involve the general public.

We have concluded that the arrangements made by the applicant to cope with the possible consequences of accidents at the site are both reasonable and prudent, and that there is adequate assurance that such arrangements will be satisfactorily implemented in the unlikely event that they are needed.

6.0 CONDUCT OF OPERATIONS AND TECHNICAL QUALIFICATIONS

Responsibility for safe operation of Units 1, 2 and 3 is vested in the Station Superintendent. He reports through the Generating Stations Superintendent and the Manager of Power Production to the President of Commonwealth Edison. The operations staff and the engineering, electrical, maintenance, instrumentation, fuel handling and clerical support groups report to the Plant Superintendent through the Assistant Plant Superintendent.

Within the onsite operating organization, responsibility for day-to-day operation of the facility rests with the Operating Engineer who reports to the Plant Superintendent. The shift complement for Units 1 and 2 will consist of 10 men. The Shift Engineer, who will be licensed as a Senior Reactor Operator (SRO), will be in charge of the crew which will include: a Shift Foreman (SRO); a Senior Control Operator (SRO); two Control Operators licensed as Reactor Operators (RO); an Equipment Operator and four Equipment Attendants. When Unit 3 is placed in operation, an additional Control Operator and Equipment Operator will be added.

The qualifications of individuals initially proposed to fill professional and semi-professional positions in the onsite operating organization have been described in the Safety Analysis Report. The minimum qualifications for these functional positions are described in the Technical Specifications. We have examined the qualifications of the incumbents and, subject to satisfactory completion of necessary AEC examinations for Operator licenses, we have concluded that the professional staff is technically competent to operate the facility.

During initial plant operation engineering support will be provided to the Dresden Nuclear Station by a plant technical staff as well as by the prime contractor and supplier of the nuclear steam supply system, General Electric, and by consultant firms. The staff is familiar with the plant and is capable of handling the preparation and review of design changes and plant modifications originating at the Dresden site.

General Electric will participate in the startup and initial operation of the plant and will continue to make available technical support to the Commonwealth Edison staff throughout the operating lifetime of the facility. On these bases, we conclude that adequate engineering capability will be available to support the applicant's operating staff.

The applicant proposes to use a two-level committee structure to perform review and audit of plant operation. The first of these committees, the Station Review Board, which is comprised of the senior members of the onsite staff, acts in an advisory capacity to the Station Superintendent. Independent audit of plant operation is provided by the Nuclear Review Board. The responsibilities and authorities for these committees are delineated in the Technical Specifications. We conclude that the review and audit structure proposed by the applicant is satisfactory.

Based on the above considerations, we conclude that the applicant is technically qualified to operate the plant and has established effective means for continuing review, evaluation, and improvement of plant operational safety.

7.0 TECHNICAL SPECIFICATIONS

The applicant's proposed Technical Specifications were presented in Amendment 19. We have reviewed these proposed Technical Specifications in detail and have held numerous meetings with the applicant to discuss their contents. Modifications to the proposed Technical Specifications submitted by the applicant were made to more clearly describe the allowed conditions for plant operation. The finally approved Technical Specifications are appended to the proposed provisional operating license. Included are sections covering safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features and administrative controls. Based upon our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of 10 CFR Part 20 limits. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features will be available in the event of malfunctions within the plant.

8.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

As noted previously, the ACRS has reviewed the application for provisional operating licenses as it applies to Unit 2 of the Dresden Nuclear Power Station, and will consider Unit 3 when its construction is nearing completion. The Committee completed its review of Unit 2 during its 113th meeting held September 4-6, 1969. A copy of the ACRS letter, dated September 10, 1969, is attached as Appendix A.

The ACRS, in its letter, made several recommendations and noted several items to be resolved by the applicant and the staff either before plant operation or on an acceptable time scale subsequent to initial operation. These items have been considered in our evaluation and include: additional seismic analyses (discussed in Section 2.4), containment inerting (Section 3.3.6) reduced steam line valve leakage effects (Section 3.3.4), modification of the auto-relief system (Section 3.5.3), means for preventing common failure modes from negating scram action and features for coping with failure to scram during anticipated transients (Section 3.5.1), separation of standby gas treatment components (Section 3.5.4), and possible effects of radiolysis of water in the unlikely event of a loss-of-coolant accident (Section 4.1).

The applicant has agreed to implement the recommendations of the ACRS. We will follow implementation of the recommendations of the ACRS during operation of the facility under the 18-month term of the provisional operating license. The ACRS concluded in its letter that if due regard is given to the items mentioned above, Unit 2 can be operated at power levels up to 2527 Mwt without undue risk to the health and safety of the public.

9.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted would be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are American citizens.

The applicant is not owned, dominated or controlled by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

10.0 CONCLUSION

Based upon our review of the application as presented and discussed in this evaluation and the report of the Advisory Committee on Reactor Safeguards, we have concluded that Unit 2 of the Dresden Nuclear Power Station can be operated as proposed without endangering the health and safety of the public.

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Date: October 17, 1969

APPENDIX A

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 10, 1969

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON DRESDEN NUCLEAR POWER STATION UNIT 2

Dear Dr. Seaborg:

During its 113th meeting, September 4-6, 1969, the Advisory Committee on Reactor Safeguards completed its review of the application by the Commonwealth Edison Company for a license to operate Unit 2 of the Dresden Nuclear Power Station at power levels up to 2527 MW(t); the Committee's review for construction was based on a design power of 2255 MW(t). The Committee had previously met with the applicant for a partial review of the application during its 110th meeting, June 5-7, 1969, and its 111th meeting, July 10-12, 1969. Subcommittee meetings with the applicant were held on May 27 and 28, 1969, at the site, and on August 21, 1969, in Washington, D. C. In the course of the review, the Committee had the benefit of discussions with the applicant, the General Electric Company, Sargent and Lundy, Incorporated, and their consultants; of discussions with the AEC Regulatory Staff; and of the documents listed. Other nuclear facilities at the site are Dresden Unit 1, which has been in operation since October 1959, and Dresden Unit 3, which is similar to Unit 2 and is in an advanced stage of construction. The General Electric Company's Midwest Fuel Recovery Plant is under construction at a separate adjacent site.

The application covers Units 2 and 3, but this report applies to Unit 2 only. The application as it applies to Unit 3 will be reviewed when its construction is nearing completion. The two units are in most respects identical, but some facilities and services are shared by Units 2 and 3, and some also by Units 1, 2, and 3. The Committee has reviewed possible interaction among units, and also the temporary arrangements necessitated by operation of Unit 2 while Unit 3 is still under construction. It is believed that the physical measures and administrative procedures to isolate the operating units from construction activities, and to provide all safety associated services to the operating units, are adequate.

Dresden Unit 2 incorporates important developments since the design of previously licensed boiling water reactors. The developments include use of jet pumps inside the vessel with an external primary recirculation system of reduced size, improvements in engineered safety features, and increased power density.

The Committee reported to you on the construction permit application for this Unit on November 24, 1965. In its report, the Committee referred to the extensive development program being conducted by the General Electric Company to substantiate the design basis of several features, including jet pump monitoring and system stability, metal-water reactions, instrumentation, and blow-down and emergency cooling. The Committee also recommended that special attention be given to other features of the design. Further recommendations applicable to Unit 2 were contained in the Committee's report of August 16, 1966, on the application for a construction permit for Dresden Unit 3. The Committee is satisfied that proper attention has been given to these matters -- additional verification of some items will be obtained during pre-operational testing and the initial operation at power.

Many improvements in safety features and procedures have evolved since the Dresden Unit 2 provisional construction permit was granted, as a result of the work of reactor suppliers, the AEC, and others. Some of these improvements have been discussed in recent ACRS construction permit and operating license reports. The applicant has agreed to incorporate several of these improvements in Dresden Unit 2. These include an improved emergency cooling system, flooding protection for the emergency cooling pumps, provision of an interlock to prevent depressurization by the automatic pressure relief subsystem if low-pressure emergency core cooling pumping capability is lost, and installation of a strong-motion seismograph.

The applicant is reviewing the seismic design of Class I structural and mechanical components of the plant and will complete his analysis before the reactor goes into operation. In the event that changes to the plant should be found necessary, such changes will be made on a time scale to be agreed upon between the applicant and the Regulatory Staff.

The Committee believes that, with the present state of knowledge of the performance of the ECCS and the course of a postulated loss-of-coolant accident, the containment should be inerted during operation of the reactor. However, it is recognized that inerting increases problems of inspecting for and repairing leaks in the primary system. It is recommended that the requirement for inerting be periodically reviewed as operating experience and further knowledge from development work currently underway are obtained, and as other means of eliminating the hazards from accident generated hydrogen are found.

Based on Dresden Unit 1 experience, the applicant stated that it will be difficult to maintain during service the very low rate of leakage through the steam line isolation valves used for accident analysis at the time of the construction permit review, and has proposed substantially larger leak rate limits than those recommended by the Regulatory Staff. The Committee believes that the leak rate limit recommended by the Staff should be met when the plant is put into operation. The Committee recommends that the applicant propose a program to ameliorate this situation and to assure the protection of the public from excessive releases of radioactivity through the closed valves in the unlikely event of an accident. This study should be completed as soon as possible, followed by necessary corrective action.

The automatic pressure relief subsystem should be modified so that at least the manual actuation of the subsystem would not be prevented by any single failure in the subsystem.

The Committee believes that, for transients having a high probability of occurrence, and for which action of a protective system or other engineered safety feature is vital to the public health and safety, an exceedingly high probability of successful action is needed. Common failure modes must be considered in ascertaining an acceptable level of protection. In the event of a turbine trip, reliance is placed on prompt control-rod scram to prevent large rises in primary system pressure. The applicant and his contractors have devoted considerable effort to provide a reliable protective system. However, systematic failures due to improper design, operation, or maintenance could obviate the scram reliability. A study is in progress on further means of preventing common failure modes from negating scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant plans to consider the results of this study and incorporate appropriate provisions in Dresden Unit 2.

Several matters are still under discussion between the applicant and the Regulatory Staff. These include review of the need for separation of redundant components of the standby gas treatment system, and final revisions to the technical specifications. The ACRS believes these matters can be resolved by the applicant and the Regulatory Staff.

Dresden Unit 2, like other reactors recently licensed for operation, has not been designed to permit the currently required high degree of accessibility for in-service inspection of the primary system boundary, including the pressure vessel and the main steam lines. The Committee believes that the proposed procedures for in-service inspection are adequate for initial operation, but believes these procedures should be reviewed at the end of a five year period to take advantage of experience in the industry and improved inspection techniques.

Continuing research is expected to enhance safety of water-cooled reactors in other areas than those mentioned, for example, by the determination of the extent of radiolytic decomposition of cooling water in the unlikely event of a loss-of-coolant accident, development of instrumentation for in-service monitoring of the pressure vessel and other parts of the primary system for vibration and detection of loose parts in the system, and evaluation of the consequences of water contamination by structural materials and coatings in a loss-of-coolant accident. As solutions to the problems develop and are evaluated by the Regulatory Staff, appropriate action should be taken by the applicant on a reasonable time scale.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, Dresden Nuclear Power Station Unit 2 can be operated at power levels up to 2527 MW(t) without undue risk to the health and safety of the public.

Additional remarks by Dr. William R. Stratton are attached.

Sincerely yours,

Stephen H. Hanauer

Stephen H. Hanauer
Chairman

Additional Remarks by Member Dr. William R. Stratton

I agree with the Committee that the applicant should be granted a license to operate the Dresden Unit 2 power plant; however, I disagree strongly with the Committee recommendation for inert atmosphere within the containment during operation of this reactor.

I take this position for the following reasons:

(1) The several accident prevention and accident limiting safeguards are sufficiently diverse and redundant to more than adequately protect the health and safety of the public in the improbable event of a very severe accident. For example, the performance of the emergency core cooling complex (sprays and flooding systems) could be severely degraded with the result that fuel pin temperatures and fission product releases would still remain within acceptable bounds. I estimate that for this reactor and site the set of safety devices is sufficient, and thus, the necessity for inerting the containment no longer exists, as may have been the case several years ago.

(2) An inert atmosphere will discourage the operating crew from entering the containment at the first opportunity in order to positively identify leaks or other abnormal phenomena detected by remote means. In the same sense, inerting would inhibit the motivation to perform routine inspections within the containment when the plant is shutdown for reasons not connected with the reactor. Thus, it is possible that the safe operation of the plant may be impeded and some degradation of equipment may occur in a manner and amount not known to the operating crew and, consequently, to management.

(3) The inerting gas is a real and present danger to anyone entering the containment even after purging is thought to have been accomplished.

For these reasons I respectfully suggest and urge the Commission not to require an inert atmosphere within the containment of the Dresden Unit 2 reactor.

References - Dresden Unit 2

1) Letter from Commonwealth Edison Company dated November 17, 1967; Volumes I and II to Safety Analysis Report.

2) Letter from Commonwealth Edison Company dated August 30, 1968; Amendments 7 and 8, Answers to AEC Questions; Volume III Proposed Technical Specifications for Dresden Unit 2.

3) Letter from Commonwealth Edison Company dated November 21, 1968; Amendments 8 and 9, Answers to AEC Questions of June 27, 1968; Volume IV to Safety Analysis Report.

4) Letter from Commonwealth Edison Company dated February 28, 1969; Amendments 9 and 10, Answers to AEC Questions of October 16, 1968.

5) Letter from Commonwealth Edison Company dated March 18, 1969; Amendments 11 and 12, Answers to AEC Questions IA and IB of January 14, 1969.

6) Letter from Commonwealth Edison Company dated April 16, 1969; Answers to Remaining AEC Questions of January 14, 1969; Answers to AEC Questions of January 22, 1969.

7) Letter from Commonwealth Edison Company dated May 20, 1969; Amendments 12 and 13 to the Application.

8) Letter from Commonwealth Edison Company dated July 2, 1969; Amendments 13 and 14, Answers to AEC Questions of May 19, 1969.

9) Letter from Commonwealth Edison Company dated July 22, 1969; Amendments 14 and 15, Answers to AEC Questions of May 19, 1969.

10) Letter from Commonwealth Edison Company dated August 5, 1969; Amendments 15 and 16 to the Application.

11) Commonwealth Edison Company's Proposed Technical Specifications and Bases for Unit 2.

12) Letter from Commonwealth Edison Company dated August 8, 1969; Amendments 16 and 17 to the Application.

13) Letter from Commonwealth Edison Company dated August 18, 1969; Amendments 17 and 18 to the Application.

14) Letter from Commonwealth Edison Company dated August 18, 1969; Amendments 18 and 19 to the Application.

References - Dresden Unit 2, Cont'd

15) Letter from Commonwealth Edison Company dated September 2, 1969; Amendments 19 and 20 to the Application.

16) Commonwealth Edison Company's Proposed Technical Specifications and Bases for Dresden Unit 2.

17) Letter from Commonwealth Edison Company dated September 4, 1969; Additional information relative to the Application.

November 10, 1969

ERRATA SHEET

TO THE

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-237

The following revisions are incorporated in the subject safety evaluation:

(1) Section 3.1.2

Change the fifth and sixth sentences in the third paragraph on page 5 to read:

> "Inadvertent continuous withdrawal of a single control rod until terminated by the rod block monitor results in a MCHFR of approximately 1.6. For the worst transient, the resulting heat generation rates are increased to approximately 21 kw/ft."

(2) Section 6.0

Change the third and fourth sentences in the first paragraph on page 30 to read:

> "The Shift Engineer, who will be licensed as a Senior Reactor Operator (SRO), will be in charge of the crew which will include: a Shift Foreman (SRO), three Nuclear Station Operators licensed as Reactor Operators (RO), an Equipment Operator, three Equipment Attendants, and a Radiation Protection Technician. When Unit 3 is placed in operation, an additional Nuclear Station Operator and Equipment Attendant will be added."

These revisions do not affect the conclusions of the AEC regulatory staff as stated in Section 10.0 of the safety evaluation.