November 12, 1970

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

CONSUMERS POWER COMPANY

MIDLAND PLANT UNITS 1 & 2

DOCKET NOS. 50-329 AND 50-330

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.0 INTRODUCTION

On January 13, 1969, the Consumers Power Company (applicant) applied to the Atomic Energy Commission (AEC or Commission) for licenses to construct and operate two pressurized water nuclear reactors to be located at the Midland site on the southern boundary of Midland, Michigan, on the right bank of the Tittabawassee River.

The combined output of the two units will be 1300 megawatts electrical (MWe) and 4,050,000 pounds per hour of process steam. The process steam will be piped to the adjacent Dow Chemical Company (Dow) plant, where it will be used in the production of chemicals.

Each of the proposed reactors is designed to operate initially at a power level of 2452 megawatts thermal (MWt) with an expected ultimate power level of 2552 MWt. The design of the engineered safety features, including the containment structures and the emergency core cooling systems, and the consequences of certain postulated accidents, have been analyzed by the applicant and evaluated by the regulatory staff for a power level of 2552 MWt. We have evaluated the thermal and hydraulic and nuclear characteristics of the reactor core at the initial power level of 2452 MWt. Before operation at any power level above 2452 MWt is authorized, the regulatory staff will perform a safety evaluation to assure that the core can be operated safely at the higher power level.

The applicant will own the proposed facilities and will be responsible for their design and construction. The nuclear steam supply systems will be furnished by the Babcock & Wilcox Company (B&W). The architect-engineer for the remainder of the plant will be the Bechtel Corporation (Bechtel). The plant will be constructed by the Bechtel Company.

Our technical safety review of the proposed plant has been based on the Preliminary Safety Analysis Report (PSAR), including Amendments 1-18, thereto. The technical evaluation of the preliminary design of the plant was accomplished by the Division of Reactor Licensing with the assistance of various consultants. In the course of our review, a number of meetings were held with representatives of the applicant to discuss the proposed plant, and we raised a number of questions which resulted in amendments to the application. A chronology of meetings and principal correspondence regarding the application is attached as Appendix A.

The Commission's Advisory Committee on Reactor Safeguards (ACRS) also reviewed the application and the 18 amendments thereto. Copies of the ACRS reports to the Commission on the Midland Nuclear Plant are attached as Appendix B.

On May 28, 1970, the applicant requested that the Commission grant an exemption pursuant to 10 CFR 50.12 of the Commission's regulations, from the provisions of 10 CFR 50.10(b) to permit construction of portions of the substructure of the auxiliary building, and the tendon galleries and foundations for the containment structures, prior to the issuance of a construction permit. The design of the applicable portions of the plant was evaluated and the exemption request was granted on July 30, 1970. The applicant was advised when the exemption was granted that it would have no bearing upon the subsequent granting or denial of a construction permit, and that work performed pursuant to the exemption would be performed entirely at the applicant's risk.

The review and evaluation of the proposed design of the facility for a construction permit is only the first stage of a continuing review by the Atomic Energy Commission's regulatory staff of the design, construction, and operating features of the Midland plant. Prior to issuance of operating licenses, we will review the final design to determine that all of the Commission's safety requirements have been met. The facility would then be operated only in accordance with the terms of the operating license and the Commission's regulations, under the continued surveillance of the Commission's regulatory staff.

2.0 FACILITY DESCRIPTION

In the Midland Nuclear Plant, the reactor, steam generators, primary coolant system, and pressurizer for each unit are housed inside their respective prestressed concrete reactor containment structures. The auxiliary building is common to the two units and houses the waste treatment facilities, components of the engineered safety features, various related auxiliary systems, and the fuel handling facilities consisting of the spent fuel storage pool and the new fuel storage facility. A separate turbine building houses the power conversion equipment for both units.

The Midland units will each employ a pressurized water reactor. Each reactor is fueled with slightly enriched uranium dioxide in the form of ceramic pellets contained in Zircaloy tubes.

Water serves as both the moderator and the coolant. Boric acid dissolved in the coolant is used as a neutron absorber to provide long-term reactivity control. Short-term reactivity control and reactivity shutdown capability are provided by top entry control rod assemblies, which are moved vertically within the core by individual control rod drives. These rods are moved in three banks, with the rods in each bank located symmetrically throughout the core.

Four reactor coolant pumps circulate the borated water through the reactor vessel and the core. The heated water then flows through two steam generators where heat is transferred to the secondary (steam) system. The reactor coolant system water then flows back to the pumps to repeat the cycle.

The secondary system steam produced in the steam generators is used in the turbine generator. In addition, secondary steam, drawn from the main steamline and from the moisture-separators between the high pressure and low pressure stages of the turbine, is passed through intermediate heat exchangers where an additional supply of water is boiled to generate 4,050,000 pounds per hour of process steam. This steam is piped to the Dow Chemical Company for use as a source of thermal energy.

The low temperature steam leaving the low pressure stages of the turbine and the intermediate heat exchangers will be condensed and the heat from this source will be discharged to the atmosphere via a large cooling pond which is located immediately adjacent to the reactor buildings.

The space for control rods in 16 of the fuel assemblies not equipped with control rod assemblies, will contain fixed burnable poison rods. These rods are located symmetrically throughout the core and are installed to assure that the moderator temperature coefficient of reactivity will not be positive during the life of the core.

A reactor protection system is provided that automatically initiates appropriate action whenever a plant condition monitored by the system approaches pre-established limits. The reactor protection system acts to shutdown the reactor, close isolation valves, and initiate operation of the engineered safety features, should any or all of these actions be required.

Redundant and independent emergency core cooling systems are provided to maintain reactor cooling, and to provide containment cooling in the unlikely event of an accident.

3.0 SITE AND ENVIRONMENT

3.1 Site Description

The proposed site is located on the right bank of the Tittabawassee River, south of and adjacent to the Dow industrial complex. It is directly south of the City of Midland. The site will include an 880 acre cooling and storage pond which will be used to reject waste heat to the environment.

The population distribution in the vicinity of the Midland plant in 1968 including both residential and business populations, is presented in Table 3.1.

TABLE 3.1

CUMULATIVE POPULATION IN THE VICINITY OF THE MIDLAND PLANT (1968)

Distance (miles)	Cumulative Residential Population	Cumulative Business Population	Cumulative Total
0-1	38	2,145	2,183
0-2	4,577	15,258	19,835
0-3	21,009	27,559	48,568
0-4	34,589	31,171	65,760
0-5	40,861	33,843	74,704

Since some persons both reside and work in close proximity to the plant, the cumulative total population figures indicated above may overestimate the total population near the site. Beyond a distance of approximately five miles, the population distribution is typical of that associated with the general agricultural utilization of the land.

In this regard, the applicant estimates that in 1965, the residential population was 54,734 within 10 miles of the proposed site, and was 243,643 within 20 miles.

For the Midland plant, the nearest boundary of the exclusion area proposed by the applicant is 0.31 miles (500 meters). The land within the exclusion area will be under the control of the applicant even though within this distance is a fenced-in waste treatment pond area now under the control of the Dow Chemical Company. Dow employees visit this area only occasionally, and no Dow employee is stationed there on a full-time basis. The applicant will be cognizant at all times of the persons within that portion of the Dow property which falls within the exclusion area. Dow and the applicant have agreed that the applicant shall have the right to remove persons from this Dow property, should a condition arise which warrants such removal.

The low population zone proposed by the applicant has an outer boundary of approximately one mile (1600 meters). The residential population within this zone is 38. The business population within this zone, predominantly employees of Dow, is 2145.

The exclusion area and the low population zone proposed by the applicant are acceptable because (1) as discussed in Section 12 of this evaluation, the calculated radiation doses at the outer boundaries of the exclusion area and the low population zone that might result from postulated design basis accidents are within the guideline doses specified in 10 CFR Part 100, (2) the residential population within the low population zone is very small, and (3) Dow has a well-structured evacuation plan available for use in the event of an emergency. The evacuation plan is discussed in Section 13.3 of this evaluation.

The guideline population center distance determined under 10 CFR Part 100 would be 1-1/3 miles. In this regard, we note that the distance to the nearest corporate boundary of the City of Midland is 0.21 miles; however, that portion of the City of Midland within 1-1/3 miles of the facility consists almost entirely of the Dow complex. Because most of the population in this area consists of employees of Dow who are subject to the evacuation plan discussed in Section 13.3 of this evaluation, we find the site acceptable.

3.2 Meteorology

Because the east-central part of Michigan where Midland is situated is in flat terrain, atmospheric flow is largely governed by large-scale continental pressure patterns. In winter, frequent storm tracks pass through the area and the ventilation rate is high and atmospheric diffusion relatively good.

Measured meteorological data are available from two wind stations at Dow, which are located about 1-1/2 miles to the northwest of the site, and from the Saginaw Tri-City Airport about 8 miles to the southeast. The applicant has based his proposed diffusion model on the data from the Tri-City Airport, correlated with measured data from the Dow facility.

The technique used by the applicant to characterize the weather data produces data in the form of gross frequency distributions rather than joint frequency distributions between stability, wind speed, and wind direction. Based on the data available, we have concluded that the available meteorological information presented by the applicant does not justify his proposed departure from the standard meteorological model we use to determine the two-hour and 30-day diffusion characteristics for accident evaluations. In our accident evaluations (see Section 11) we have used our standard model to provide a conservative basis for accident evaluations in the absence of adequate local meteorological data.

The applicant has agreed to conduct an onsite meteorological measurements program to include (1) continuous time-history measurements of wind velocity and direction at an elevation of 100 feet above the general terrain and (2) differential temperature measurements made at the 10 foot and 90 foot levels where the wind data are obtained. A minimum of one year's data will be available prior to our review of this plant for an operating license. Based on our evaluation of the proposed program, we and our meteorological consultant, the Environmental Science Services Administration (ESSA), conclude that the measurements proposed will be adequate to determine the diffusion characteristics of the site. The report of ESSA is attached as Appendix C.

The applicant will design vital structures to withstand the combined loads resulting from a tornado having a uniform tangential wind speed of 300 miles per hour, a translational wind speed of 60 mph and a differential pressure drop of 3 psi in 3 seconds. These wind speeds and this pressure drop are consistent with our conservative estimates of the characteristics of tornadoes in the eastern and midwestern parts of the United States and thus we conclude that the use of these assumptions is acceptable for the design of the Midland site to withstand the effects of tornadoes.

3.3 Geology and Seismology

The site is located on a glacial lake plain. Bedrock, consisting of shales interbedded with sandstones and siltstones, is located 350 to 360 feet below ground level. The glacial drift overlying the bedrock consists of a thin upper layer of sand, zero to 40 feet in thickness, which is clayey in some areas, followed by a zone of compact impermeable clay. This clay layer is 130 to 190 feet thick. All heavy plant structures are founded in this zone. The clay zone is followed by layers of sand and gravel to bedrock.

Dow is engaged in solution mining of sodium chloride in the vicinity of the site, at depths of 4100 to 4300 feet. Dow has also conducted brine extraction operations at a depth of 5100 feet. The applicant has stated that no future salt mining operations will be conducted under or immediately adjacent to the plant site area. The applicant has calculated that the maximum subsidence at the site that might result from these operations, considering the superimposed effects of all wells in the vicinity of the site, is 0.36 inch with a slope across the plant site of 0.02 inch per 100 feet. Considering that the plant structures can safely withstand a uniform slope of one inch per 100 feet across the site, the applicant concludes that the structural design criteria would not be violated if subsidence should occur.

Based on our review of the material submitted by the applicant, we have concluded that the potential for subsidence at this site is very low. Dow has an established grid of benchmarks in the vicinity of their salt wells and has conducted observations of these benchmarks since 1958. These data indicate no evidence of surface subsidence, nor is there any suggestion of a trend toward surface subsidence in the accumulated 11-year record. We have concluded that continued surveillance for subsidence should be maintained throughout the life of the plant to permit evaluation and corrective action, if necessary, if subsidence does occur.

The applicant has agreed to establish a more extensive and more accurate surveillance record prior to operation of the Midland plant. The details of this expanded surveillance program will be developed by the applicant and submitted for our review during construction of the plant. Our consultant, the U. S. Geological Survey, concurs in these conclusions, and its report is attached as Appendix D.

The seismicity of the site has been evaluated by the U. S. Coast and Geodetic Survey (USC&GS). The USC&GS recommends, and we concur, that the seismic design accelerations for the Operational Basis Earthquake* and the Design Basis Earthquake** should be 0.06g and 0.12g, respectively. The applicant has agreed to design to these accelerations. A strong motion accelerograph will be installed in the facility to provide information on the seismic accelerations experienced at the site in the event of an earthquake. The actual system employed, its location, and the requirements for its use will be determined at the operating license stage of our review. We find this to be acceptable. The report of the USC&GS is attached as Appendix E.

3.4 Hydrology and Flood Protection

The Midland plant site is located on the right bank of the Tittabawassee River. River water downstream of the plant is used only for industrial cooling purposes. The flow rate of the Tittabawassee River is low. For this reason, a cooling pond containing 7900 acre-feet of useful storage volume is provided to permit continued plant operation without withdrawal of water from the river for cooling purposes. Water will be withdrawn from the river to replenish the cooling pond only if the river flow rate is above 350 cubic feet per second.

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*The Operational Basis Earthquake for a reactor site is one which causes that vibratory ground motion for which all features of the facility necessary to permit continued operation are designed to remain functional.

**The Design Basis Earthquake for a reactor site is one which causes that vibratory ground motion for which all features of the facility necessary to protect the health and safety of the public are designed to remain functional.

The nearest municipal water supply identified by the applicant which could be affected by release of radioactive effluents from the Midland plant is located on Saginaw Bay, 40 to 50 miles from the site. Our evaluation of the effects of releases of radioactive material on these municipal water intakes is given in Section 9 of this evaluation.

The applicant will determine the probable maximum flood level at the site using calculational techniques that we have evaluated and found to be acceptable. The applicant will design vital structures to withstand the effects of the probable maximum flood level so calculated. We find this criterion to be acceptable. The techniques used in calculating the probable maximum flood and the general hydrology of the site have been reviewed and found acceptable by the U. S. Geological Survey. The USGS report is attached as Appendix D. We will review the applicant's calculation of the probable maximum flood level during construction of the plant to assure that the calculational techniques have been properly employed.

3.5 Environmental Monitoring

A preoperational environmental radiation survey program will be conducted at the Midland site by the applicant. This program will consist of analyses of six air particulate samples weekly, six measurements of radioactive iodine activity in the air weekly, three measurements of the gross beta activity of the waters of the Tittabawassee River and Chippewa Rivers monthly, three measurements of the tritium content of the waters of the

Tittabawassee and Chippewa Rivers monthly, and nine measurements of the gamma activity of samples of fish and other aquatic life monthly, when possible. An expanded monitoring program will continue during operation of the plant. We have evaluated the Midland plant pre-operational monitoring program relative to the number, type, and location of the sampling stations and the analyses performed and conclude that the program will provide a valid basis for evaluating the radiological impact of the plant on the environs by comparing the future levels of radioactivity with the preoperational levels. We will require that the preoperational monitoring program be in operation at least two years prior to initial criticality.

The Fish and Wildlife Service, U. S. Department of the Interior (F&WS), reviewed the application and made certain recommendations with respect to radiological monitoring by the applicant. The F&WS report, a copy of which is attached as Appendix F, has been transmitted to the applicant. We have urged that the applicant follow the F&WS recommendations.

3.6 Accidents at Dow Chemical Company

The applicant has evaluated the potential effects on the Midland plant of accidents that might occur at the adjacent Dow Chemical Plant. The applicant has stated that all of the Dow units that present a significant explosive hazards are located at least one mile from the reactor plant and that none of the potential accidents would cause measurable damage at distances greater than 1,000 feet from the processing unit involved.

Large quantities of toxic chemicals are stored at the Dow plant. Dow has identified chlorine, bromine and methyl bromide as those chemicals representing the maximum toxic hazard at the Midland plant. Of these, the applicant has indicated that chlorine presents the greatest hazard. The applicant has evaluated the effect on the Midland Reactor Plant site of the chlorine release at Dow that might result from the massive failure of a liquid chlorine storage tank located approximately 1.5 miles from the reactor. This tank, a 44-foot-diameter sphere supported above ground on eight legs, contains approximately 2,000 tons of liquid chlorine at atmospheric pressure. The tank and its supporting structure are designed so that in the event of a massive failure, the liquid chlorine would drop through an opening into a covered concrete containment pit located beneath the tank. The containment pit is surrounded by a dike to contain the chlorine and to direct its flow into the pit. A sump pump located in the pit permits pumping the chlorine in the pit to tank cars. In addition, the atmosphere in the pit would be vented to a caustic scrubber using a vent fan. Both the sump pump and the vent fan can be operated on emergency diesel power.

The applicant has calculated the chlorine concentrations in the Midland Reactor Control Room that might result from release of the contents of the chlorine storage tank to the containment pit in approximately 12 minutes assuming (1) a wind speed toward the Midland plant of 1 meter per second with Pasquill Type F diffusion conditions, (2) failure of the caustic scrubber, (3) air infiltration into the control room at a rate of 16 cubic feet per minute, and (4) filtration of the air being recirculated by the control room ventilation system by an impregnated charcoal filter with an efficiency for chlorine removal of 99 percent. The calculations indicate that a peak chlorine concentration in the control room of 3.6 parts per million would occur approximately 9 minutes after arrival of the cloud at the reactor site (49 minutes after the accident). The applicant has stated that the chlorine concentration in the control room could be reduced to less than 1 part per million in approximately 90 minutes by the use of the filtered ventilation recirculation system.

The threshold limit value (TLV) established by the American Conference of Governmental Industrial Hygienists for continuous 8-hour exposure by industrial workers is 1.0 part per million for chlorine, 0.1 part per million for bromine, and 20 parts per million for methyl bromide. In order to assure that the operators can operate with full effectiveness in the event of a release of a toxic chemical at Dow without relying on respiratory equipment, we will require that the control room will be designed to limit the concentration to less than the TLV at all times following a release at Dow.

4.0 REACTOR DESIGN

The Midland Reactors will operate at core power levels up to 2452 MWt, and will have an ultimate power level of 2552 MWt. All core physics, thermal and hydraulic characteristics have been evaluated for the 2452 MWt power level. The proposed power level and mechanical design of Midland Reactors are the same as those of Oconee Nuclear Station Units 1, 2 and 3. On the basis of our previous review of these plants, and upon our subsequent review of Babcock and Wilcox Topical Reports on related reactor core design and analysis subjects, we conclude that the Midland plant design is acceptable with regard to core physics, core thermal and hydraulic design, and core mechanical design.

During plant operation, changes in the core power level or in the control rod configuration can cause time-dependent variations in local power distribution as a result of variations in the concentration of fission products and their radioactive decay products. The most significant fission product-decay product chain with regard to core behavior is the decay of iodine-135 to xenon-135, since the latter is a strong absorber of thermal neutrons.

The local oscillations in the neutron flux and in the power level can occur even though the average power level of the core is maintained constant, and the magnitude of the oscillations may decrease, remain constant, or increase with time. The applicant is performing analyses to determine the stability of such oscillations for various core configurations. Results to date indicate (1) the core will not be subject to divergent azimuthal or radial power oscillations, and (2) potential axial power oscillations can be controlled by movement of the part-length control rods. A research and development program is now underway to obtain more detailed information on the potential for oscillations. This is discussed in Section 15.0 of this evaluation.

As presently proposed, fuel clad failure, and subsequent increase of reactor coolant system activity would be detected by a process radiation monitor located in the letdown line from the reactor coolant system to the makeup and purification system. Improved means for prompt detection of fuel clad failure are under development within the industry, and as recommended by the ACRS, we will require that the applicant include in the final plant design the best equipment available to detect promptly the gross failure of a fuel element.

5.0 REACTOR COOLANT SYSTEM

5.1 General

The reactor coolant system design is similar to that reviewed and approved for the Rancho Seco plant. The reactor coolant system for each reactor will be designed to withstand normal loads of mechanical, hydraulic and thermal origin, plus anticipated seismic loads from the operational basis earthquake within the stress limits of the codes discussed below.

5.2 Design Criteria

The Midland reactor vessels will be designed and fabricated in accordance with the 1968 edition of the ASME Boiler and Pressure Vessel Code, Section III, Class A, plus the summer, 1968 addendum, and Code Cases 1332, 1335, and 1339. The vessel design is the same as that for the vessel of the Rancho Seco plant. The reactor coolant piping will be designed to the ANSI B31.7 Nuclear Power Piping Code dated February 1968, including the June 1968 errata. The proposed design criteria for the reactor vessel and piping comply with the proposed Section 50.55a (c)-(d), 10 CFR, published in the Federal Register for comment on November 25, 1969.

We have reviewed the codes, the plans for design and fabrication, and the quality specified for the reactor vessels and coolant piping and conclude that the reactor vessels and coolant piping as planned are acceptable.

5.3 Seismic Design Methods

All system components will be designed to withstand the forces that would result from the blowdown of the reactor coolant system as a result of a design basis loss-of-coolant accident, concurrent with the design basis earthquake loads.

The applicant has defined Class I (seismic) structures, systems and equipment as those whose failure could cause a release of radioactivity that would result in calculated concentrations at the site boundary in excess of 10 CFR 20 limits, or those necessary for safe shutdown of the facility. Class II (seismic) structures, systems, and equipment are those whose failure would not result in the release of radioactivity which would exceed 10 CFR 20 levels at the site boundary and would not prevent safe shutdown. We have examined the applicant's categorization of plant structures and components and consider the categorization acceptable.

We have reviewed the applicant's proposed seismic design methods for the mechanical equipment which is part of the reactor coolant system and of all other Class I (seismic) systems. For flexible equipment (that having a fundamental frequency less than 20 cycles per second), the response spectra at the points of mounting will be determined from the predicted response of the structure. For rigid equipment (that having a fundamental frequency greater than 20 cycles per second), the peak acceleration at the level of the support predicted from the structural response spectrum will be used. In addition, the quality assurance program calls for verification by the applicant of the analytical and empirical methods used by the vendor to certify that this equipment meets the specifications developed on the above bases. We find this procedure to be acceptable since it follows established design practices. Our seismic design consultant, John A. Blume and Associates, Engineers has also reviewed and accepted this design method. His report is attached as Appendix G.

5.4 Reactor Vessel Internals

The reactor vessel internals will be designed to function within the stress limit criteria of Acticle 4, Section III of the ASME Boiler and Pressure Vessel Code for normal design loads of mechanical, hydraulic, and thermal origin, and loads that would result from the operational basis earthquake and from anticipated transients. All internal components will be designed to withstand the loads which will result from the combined design basis earthquake and loss-of-coolant accident. The strain limits for the material under this combined load will be held to less than 20% of the ultimate strain for this material (this corresponds to a stress limit of approximately 2/3 of the ultimate stress). All welds necessary to maintain the structural integrity of the core support structure will be performed by operators qualified in using procedures in accordance with Section IX of the ASME Boiler and Pressure Vessel Code and inspected to the acceptance requirements of Section III of this same code. We find these design limits and procedures acceptable since they follow established design practices.

Major core and core support components have been analyzed to provide assurance that they are not vulnerable to vibratory excitation. These analyses have considered inlet flow impingement and turbulent flow. Calculations have been performed that establish that the possible natural resonant frequencies of the components are at least twice the excitation frequencies in the system. Confirmatory vibration testing will be conducted as part of the preoperational startup program. Test runs will be made with the plant at hot and cold conditions before and after fuel loading, and with all permitted pumping arrangements. Instrumentation will consist of a number of accelerometers located at various positions relative to the core internals. The total measured accelerations and the deflection data obtained will be analyzed to determine the amplitudes and frequencies of the total response of the structures. We have concluded that the analytical effort and the proposed testing program is acceptable.

5.5 Protection from Missiles and Pipe Whip

The applicant has proposed to protect the primary system and all engineered safety features from damage that might be caused by missiles generated as the result of equipment failure within the containment structures. This will be accomplished either by separation of redundant systems or by the use of missile shields. In addition, the orientation of components that could generate missiles will be considered during design. Direct shielding will be provided to prevent missiles generated by the failure of pressurized components from damaging other equipment. Although the design has not progressed sufficiently to determine the potential missile sizes and masses, we have evaluated and found the missile penetration formulae and missile protection criteria proposed by the applicant consistent with established practices and AEC criteria and acceptable.

In the event of a failure of a pressurized line, the reaction forces resulting from the discharge of fluid through the rupture can cause displacement of the affected pipe (pipe whip). To prevent a whipping pipe from striking, and potentially damaging safety-related equipment, the reactor coolant system and all other Class I (seismic) items within the containment structure including the applicable portions of the emergency core cooling system, will be protected by (1) physical separation from Class II (seismic) high pressure systems, (2) separation of redundant systems and/or components, (3) restraint of lines which could whip and damage other Class I (seismic) systems. We find these criteria consistent with AEC criteria and acceptable.

5.6 Leakage Detection

Three means will be available to detect leakage from the primary system or from other systems within each of the reactor containment structures: (1) humidity detectors, (2) reactor building sump level indicators, and (3) radiation monitors within the containment structures that monitor the discharge of the air coolers. The array of leak detection instrumentation to be provided for the Midland plant is sensitive, provides timely alarms, and is redundant and diverse. On this basis, we conclude that the proposed leakage detection systems are acceptable. The limits on permissible reactor coolant system leakage rates for plant operation will be established during the preparation of technical specifications for the operating license.

5.7 In-Service Inspection

Although detailed in-service inspection plans for the reactor coolant system components have not yet been developed, the applicant will comply with the draft ASME Code for the In-Service Inspection of Nuclear Reactor Coolant Systems (N-45). This draft is equivalent to Section XI of the ASME Boiler and Pressure Vessel Code. This we find acceptable and consistent with proposed section 50.55a(f), 10 CFR, published for comment in the Federal Register on November 25, 1969.

The reactor coolant pump flywheels will be ultrasonically tested prior to initial startup. In addition, each flywheel will be inspected once in each ten-year interval by ultrasonic inspection or an equivalent method. Specific requirements for in-service inspection of the reactor coolant system, the pump flywheels, all reactor vessel supports, and of the engineered safety features will be established during the preparation of technical specifications for the operating license.

6.0 CONTAINMENT AND CLASS I (SEISMIC) STRUCTURES

6.1 Class I (Seismic) Structures Other Than Containment

Class I (seismic) structures include (1) the containment structures (discussed in Section 6.2 of the evaluation); (2) the portions of the auxiliary building housing the engineered safety features, the control room, or radioactive material; (3) the enclosures for the service water pumps, the auxiliary feedwater pumps, and the diesel generators; and (4) the diesel fuel storage facility. The design loading criteria established by the applicant for all Class I (seismic) structures, other than the containment structures, consider normal operating conditions as well as the combined loads associated with the design basis earthquake, the forces resulting from rupture of any one pipe, loads resulting from thermal gradients, and the normal live and dead loads to which the structure is subjected. We have evaluated the loading criteria proposed and find them consistent with established practices and acceptable.

The applicant has considered potential interaction between Class I (seismic) and Class II (seismic) components and structures during seismic excitation to assure that failure of a Class II (seismic) structure or component would not damage a Class I (seismic) item. In this regard, even though the turbine building is not considered a

Class I (seismic) structure, it is designed in such a manner that it will not collapse under seismic or tornado loadings. This design requirement is acceptable.

6.2 Containment Structures

6.2.1 Description

The containment structures proposed for Midland are similar to other Bechtel-designed PWR containments such as Arkansas Nuclear One Unit 1 in that they are prestressed concrete cylinders and domes supported on reinforced concrete foundation slabs. The Midland containment structures are founded on stiff cohesive soil.

Lines penetrating the containment structure are equipped with double isolation barriers such that no single failure or component malfunction can result in leakage from the containment structure to the atmosphere. These barriers consist of closed piping systems and isolation valves, where applicable. Isolation valves are automatically closed in the event high containment pressure is detected. To further reduce leakage, the applicant has agreed to install pressurized weld channels, or their equivalent, over the seam welds in the liner and to provide an isolation valve seal water system and a penetration pressurization system.

6.2.2 Containment Functional Evaluation

We have investigated the transient pressure that might be produced in the containment in the event of a design basis loss-of-coolant accident. Various loss-of-coolant containment pressure transients were investigated by the applicant using the Bechtel developed COPATTA computer code and the CONTEMPT code, developed by the Idaho Nuclear Corporation. The applicant has calculated a peak containment pressure of 60.0 psig using these codes. We have performed independent calculations using the CONTEMPT computer code and our results agree with those obtained by the applicant. In addition, we have determined that the estimated ratio of the surface area available for heat transfer to the containment free volume for the Midland plant is consistent with that estimated for other facilities. The design pressure for the containment structures is 67.0 psig, which exceeds the peak calculated pressure in the containment by more than 10%. We conclude that this margin is adequate to cover possible uncertainties and that the design pressure for the containment structures is acceptable.

6.2.3 Containment Structural Evaluation

The containment liner will be welded 1/4 inch steel plate conforming to ASTM A-285 Grade A firebox quality with a minimum yield strength of 24,000 psi and a minimum elongation in an 8-inch specimen of 27%. The concrete will utilize Type II cement and will have a 28-day compressive strength of at least 5,000 psi in the containment walls and dome, and 4,000 psi in the reinforced foundation mat. Reinforcing steel in the base mat and around penetrations will conform to ASTM A-615-68, Grade 60, while the rest will be Grade 40. Bars larger than No. 11 will be spliced by the Cadweld process, in accordance with strength and testing criteria that we find acceptable. These materials and specifications are consistent with current design practice and are acceptable.

The proposed prestressing system is the same as that which we have reviewed and found acceptable for the Arkansas Nuclear One Unit 1 facility.

We have evaluated the proposed design loads, load combinations, acceptance limits and design techniques to be used for the design of the plant for normal operation, for accident conditions and for design basis environmental conditions due to earthquakes, tornadoes, and flooding. For the seismic design, the combined stresses will remain within the allowable limits specified in the applicable structural design codes even when the calculated seismic stresses are increased by 50% for ground motions in the period range from 0.2 to 0.6 seconds using the Housner spectra. The report of our design consultant, John A. Blume and Associates, Engineers, is attached as Appendix G. We and our seismic design consultant have concluded that the proposed design is acceptable.

6.2.4 Testing and Surveillance

A pre-operational proof test of the containment structure at 80 psig (119.4% of design pressure) and leak-rate tests at 67 psig (design pressure) and several lower pressures will be performed prior to operation of the reactors. Subsequent periodic leak-rate tests at reduced pressures will be performed. The applicant proposes to conduct periodic structural surveillance by obtaining lift-off readings on nine representative tendons. In addition, three wires of a tendon in each of three direction groups (hoop, vertical, dome) can be removed and inspected. We conclude that the pre-operational testing program is acceptable and that adequate provisions are available to conduct an acceptable post-operational testing program. The details of the post-operational testing program will be established in the technical specifications.

7.0 ENGINEERED SAFETY FEATURES

7.1 Emergency Core Cooling System

The design of the Emergency Core Cooling Systems (ECCS) proposed for the two Midland reactors is the same as that reviewed and approved for other B&W-designed plants, such as the Rancho Seco plant. No single failure of active ECCS components, and no single failure of passive components during the long-term cooling phase, will reduce system performance capability below acceptable levels. In this regard, the applicant has agreed to provide a sealed compartment surrounding the line from the containment sump to the suction of the low-head safety injection pump. In the event of a failure in the line upstream of the isolation valve, this compartment w [illegible] prevent leakage from the containment structure of either water or air. All piping for the ECCS will be designed in accordance with the ANSI B31.7 Code for Nuclear Power Piping. The ECCS for the plants will consist of the following subsystems designed to protect the core for the complete spectrum of assumed hot or cold leg break sizes:

(1) A high pressure injection system that normally operates as part of the primary make-up and purification system. Two independent and redundant systems utilize three high pressure pumps, each capable of injecting a minimum of 340 gallons per minute of borated water into the reactor coolant system.

(2) A core flooding system that automatically discharges the contents of two independent and redundant storage tanks (containing a total of 1880 cubic feet of borated water) into the reactor pressure vessel when the reactor coolant system pressure drops below approximately 600 pounds per square inch. The pressure and fluid level within these tanks will be displayed in the control room and alarms will sound for any abnormal condition.

(3) A low pressure injection system that normally operates as a portion of the decay heat removal system and consists of two independent and redundant systems, each capable of injecting 3000 gallons per minute of borated water into the reactor vessel.

The source of coolant for both the ECCS high pressure injection and low pressure injection subsystems is a 650,000 gallon borated water storage tank. The level of coolant in this tank will be displayed in the control room, and alarms will sound for any abnormal condition. The concentration of boron in all emergency injection coolant systems will be sampled and analyzed periodically to assure that the boron concentration is maintained at or above 2270 parts per million.

The emergency core cooling system is designed to limit the maximum fuel clad temperature in the event of a loss-of-coolant accident to less than 2300°F for any size primary system pipe rupture up to and including the double-ended rupture of the 36-inch diameter outlet pipe. In analyzing the core thermal transient following the loss-of-coolant accident, it is assumed that only the core flooding tanks, one high pressure injection pump, and one low pressure injection pump provide coolant to the core. Delivery of the coolant by the low pressure injection pump is assumed not to start until the primary system pressure has been reduced to 100 psi or after 25 seconds, whichever occurs later.

The applicant has calculated the maximum fuel clad temperatures for a spectrum of hot leg and cold leg break sizes using a modified version of the FLASH I computer code. This code describes the reactor coolant system by the use of two control volumes for the primary loops on the basis of temperature distribution and one control volume for the pressurizer. Resistances to flow are calculated by dividing the reactor coolant system into 24 regions and calculating the volume-weighted flow resistance for a given rupture location based on normal flow resistances. The model incorporates a variable velocity steam bubble rise model.

The highest cladding temperature calculated is 2007°F. This temperature results from the assumed rupture of a 36-inch diameter hot leg pipe. Prior to installation of equipment for the emergency core cooling system, we will require that the applicant verify the results of his analyses using more sophisticated multi-node analytical techniques which represent the reactor coolant system by the use of several control volumes, rather than the two used in the present calculational technique. In addition, the code used in the verification of the performance of the emergency core cooling system will utilize the data available from the appropriate research and development programs discussed in Section 15 of this evaluation. We have concluded that the applicant's preliminary design and the analysis effort to be performed are acceptable.

As with our previous reviews, we will require that the ECCS (1) limit the peak clad temperature to well below the clad melting temperature, (2) limit the fuel clad-water reaction to less than one percent of the total clad mass, (3) terminate the clad temperature transient before the geometry necessary for core cooling is lost, and before the clad is so embrittled as to fail upon quenching, and (4) reduce the core temperature and then maintain core and coolant temperature levels in the subcooled condition until accident recovery operations can be accomplished. The ECCS will provide this protection for all pipe breaks up to and including the double-ended rupture of the largest reactor coolant pipe.

7.2 Iodine Removal

The applicant will provide an iodine removal containment spray system for Midland similar to that proposed and approved for Arkansas Nuclear One Unit 1. In order to increase the iodine removal effectiveness of the spray, the Midland design will inject an alkaline sodium thiosulfate solution into the borated water sprayed into the containment by each of the two independent 1300 gpm containment spray systems. Sodium thiosulfate and sodium hydroxide are added to the system by separate redundant metering pumps. During the initial spray phase when spray water is drawn from the storage tanks, the spray solution in the containment will be alkaline and will not exceed a pH of 11. After mixing is complete, the initial composition of the mixture of spray water, emergency core cooling system water, and reactor coolant system water will have a pH of approximately 9. The spray system will be designed in such a manner that adverse pH conditions cannot develop to the extent that they will significantly affect system performance.

In evaluating the iodine removal effectiveness of the chemical additive spray system, we have used a more conservative calculational model than that used by the applicant. Our results predict a spray removal constant of 2.5 hours^{-1}. Our evaluation of the radiological consequences of a loss-of-coolant accident, presented in Section 12 of this evaluation, is based on the use of this value for the spray removal constant.

Research and development effort is being conducted on the long-term stability of the alkaline sodium thiosulfate solution under post-loss-of-coolant accident conditions, and on the material compatibility aspects of the spray solution with all exposed construction materials. This program is described in Section 15 of this evaluation. In view of this R&D program and since offsite doses calculated using our conservative assumptions are within 10 CFR Part 100 guideline values, we find the iodine removal equipment acceptable.

7.3 Containment Heat Removal Systems

Containment heat removal following a loss-of-coolant accident can be achieved by the use of either the containment spray system, or the fan-cooler system, or by use of portions of both systems. Each system is capable of initially removing 200,000,000 Btu per hour from the containment atmosphere at design conditions. The applicant has calculated the effectiveness of the containment heat removal system assuming that only one of the two containment spray pumps and two of the four fan-cooler units operate. For these conservative assumptions, the containment pressure would decrease rapidly from the peak pressure reached during the transient and would reach a pressure of 11 psig in 52 minutes. At this time the spray water available in the borated water storage tank would be depleted and recirculation of water from the containment sump to the spray headers would be initiated. Since the temperature of the water in the containment sump would be higher than the temperature of the containment building atmosphere at this time, the initiation of the recirculation phase of the containment spray system would cause the pressure in the containment structure to rise to 17 psig at 97 minutes. The temperature of the water in the containment sump soon would drop below the temperature of the containment atmosphere, because of the cooling action of the decay heat removal system, causing the containment pressure to decrease to a pressure of 10 psig in 6.7 hours and to a pressure of 4 psig in 27.8 hours. Because the containment heat removal systems would cause the containment pressure to drop to a low value within the first day following a loss-of-coolant accident, we conclude that the capacity of the containment heat removal systems proposed is adequate.

7.4 Post-Accident Hydrogen Control

In the event of a loss-of-coolant accident, radiation from the core and from fission products which have escaped from the core will dissociate some of the cooling water into gaseous hydrogen and oxygen. In addition, hydrogen is produced by chemical reactions between the alkaline spray solution and metals in the containment, and by any metal-water reaction that might occur in the core as a consequence of the loss-of-coolant accident. Continued evolution of hydrogen could increase the concentration in the reactor containment to a point where hydrogen ignition could occur and thus provide an additional source of energy to the containment structure.

We have estimated the hydrogen buildup time and the potential radiological consequences that would ensue should it be necessary to purge the Midland containment to reduce the hydrogen concentration. We have calculated that a hydrogen concentration of 3.5 percent could be reached in nine days. Purging of the containment at this time would result in additional thyroid and whole body doses at the outer boundary of the low population zone of 54 and 3 Rem, respectively. We have concluded that purging is not acceptable as the primary means of limiting hydrogen buildup for the Midland plant, and will require the applicant to provide a method for the control of hydrogen other than purging, but that capability for purging also be maintained as a backup to the hydrogen control system. We will review the detailed design of the proposed hydrogen control system as a part of our operating license review of this plant, and will require that an acceptable hydrogen control system be provided prior to issuance of an operating license.

8.0 INSTRUMENTATION, CONTROL AND POWER SYSTEMS

8.1 Instrumentation and Control

The reactor protection system instrumentation and control systems, and the instrumentation systems which initiate and control the engineered safety features are substantially the same as those proposed and found acceptable for the Three-Mile Island Unit 2 plant. The following discussion is limited to those features of the design that differ from the Three-Mile Island Unit 2 design and to those areas where new information is available. These areas concern only the engineered safety feature instrumentation design and the requirement for a diverse engineered safety feature initiation signal.

Conformance of the protection system to the Commission's proposed General Design Criteria, as published in the Federal Register on July 11, 1967, and the Proposed IEEE Criteria for Nuclear Power Plant Protection Systems (IEEE 279) dated August 1968, served, where applicable, as the principal basis for our conclusion that the instrumentation and control system designs are acceptable.

In the Three-Mile Island Unit 2 design, three instrument channels are provided to monitor each variable required to initiate engineered safety feature action. The Midland design uses four instrument channels arranged in a two-out-of-four coincidence logic to initiate engineered safety feature action. The applicant has stated that the system will meet the requirements of the Proposed IEEE Criteria for Nuclear Power Plant Protection System (IEEE-279) dated August 1968. We have concluded that this system provides added redundancy and is acceptable.

The applicant proposes to monitor containment radiation levels and to initiate isolation of all containment penetrations open to the containment atmosphere when the radiation levels exceed predetermined limits. Four reactor building radiation monitoring instrumentation channels, arranged in a two-out-of-four coincidence logic, are provided for this function. The applicant has stated that this system will be designed to meet the requirements of IEEE-279. We have concluded that this proposed design is satisfactory.

In the Midland design, in the event of a loss-of-coolant accident the emergency coolant injection system would be actuated by either low reactor coolant system pressure or high containment pressure; however, reactor trip would be initiated only by low reactor coolant pressure. Since the analyses used to evaluate the effectiveness of the emergency core cooling system assume that a reactor trip would occur, the applicant agreed to provide a diverse signal in addition to that of low reactor coolant system pressure to trip the reactor in the event of a loss-of-coolant accident. The ACRS has recommended that this diverse reactor trip be initiated by a high containment pressure signal. We will require that the additional trip signal be provided from high containment pressure or another suitably diverse signal that can be demonstrated to be acceptable. This matter will receive additional review during final design. We conclude that the applicant's commitment is satisfactory for the construction permit stage of review.

8.2 Offsite Electrical Power Systems

The Midland plant will be interconnected to the applicant's distribution system through 345 kilovolt (kV) and 138 kV circuits. Power from the generator of each unit will be fed via separate circuits to the 345 kV switchyard. This switchyard will be interconnected to the adjacent 138 kV switchyard by means of step-down transformers. Both switchyards will be arranged in a two-bus, breaker-and-one-half configuration. Five 345 kV and six 138 kV transmission circuits emanate from their respective switchyards sharing three rights-of-way.

The applicant has designed the transmission system to minimize the probability of power failure due to faults in the electrical power system. The design criteria include the requirement that system stability be maintained in the event of the sudden outage of all generating capacity at any plant. In view of the interconnection capability and the design criteria outlined above, we conclude that the transmission system is acceptable.

Two startup transformers provide redundant, independent sources of offsite power to the 4160 volt engineered safety feature buses of Unit No. 1 and 2. One startup transformer is supplied by a 138 kV transmission circuit from the 138 kV switchyard. The second startup transformer is supplied by a 138 kV transmission circuit connected to the Dow South Substation of the Consumers Power Company. This circuit is mounted on independent towers and on a right-of-way separated from that of the circuit for the first startup transformer. Upon loss of the normal supply, each transformer is automatically connected to one of the two engineered safety feature buses in each unit. Therefore, loss of one startup transformer will result in the loss of offsite power to only one of the two redundant engineered safety feature buses in both units and will not negate the operation of the minimum engineered safety features that are required for safety.

We have concluded that because of the capacity and redundancy provided, and the relative independence of the redundant power sources, the offsite electrical power system is acceptable.

8.3 Onsite Electrical Power System

The design of the onsite electrical power system utilizes the split-bus concept. The engineered safety feature loads for each unit are divided between two 4160 volt buses such that the operation of either one will supply minimum safety requirements. Two diesel generators will be provided, each assigned to supply one of the aforementioned 4160 volt buses in each unit. Each diesel generator will be sized to provide minimum engineered safety feature loads in one unit and minimum safe shutdown loads in the other unit without exceeding the continuous rating of the diesel. The applicant's preliminary load calculations indicate that a diesel generator with a 3000 kW continuous rating is required. Test data will be supplied to confirm the suitability of this size diesel generator as an onsite emergency power source prior to the operating license review.

The redundant diesel generators and the engineered safety feature buses will be located in separate rooms of a Class I (seismic) building so that an incident involving one diesel generator or bus will not involve its redundant counterpart, either physically or electrically. Each diesel generator will be provided with a fuel tank of sufficient capacity to permit operation at full power for four hours. The main diesel fuel storage facility will have sufficient capacity to permit full power operation of a diesel generator for seven days.

Two dc systems will be provided. Each system will use two separate, redundant and independent battery supplies. One system utilizes 125 volt batteries and the second utilizes 250 volt batteries. The dc emergency loads for each unit are divided between the two 125 volt buses such that operation of either one will supply the minimum required load. One emergency bus will normally be supplied from two battery chargers, each of which is connected to a separate engineered safety feature motor control center. In addition, each battery will be located in a separate ventilated room designed to Class I (seismic) standards. The racks on which the batteries are mounted will also be designed to meet seismic requirements. These batteries will be adequate to assure a safe and orderly hot shutdown in the event that all ac power is lost. The 250 volt batteries will provide power to non-safety-related loads, such as the turbine auxiliary motors. The 250 volt system is separate, physically and electrically, from the 125 volt system.

The 120 volt ac system for the plant protection instrumentation and other essential plant controls consists of four distribution buses for each unit. Each bus is supplied through a static inverter from one of the aforementioned 125 volt dc buses. This arrangement provides four independent power sources for the protection system instrumentation of each unit. We conclude that the design of the onsite electrical power system is consistent with current practice and AEC criteria and is acceptable.

8.4 Installation Criteria

The applicant has documented his criteria for cable design, selection, and routing. These have taken into consideration protection against the loss of redundant channels of protection from a single cause such as fire, and the physical identification of safety related circuits and components. As recommended by the ACRS, the applicant will develop more detailed criteria and procedures for installation of protection and emergency power systems. We will review these criteria and procedures prior to actual installation.

8.5 Environmental Testing

The applicant has identified the instrumentation and electrical equipment, including cables located within containment that are required to operate during and subsequent to an accident. The applicant has stated that similar items have been or will be subjected to qualification tests under combined conditions of temperature, pressure, and humidity, and separately, under accident radiation doses.

Additionally, the applicant has provided seismic design criteria for the reactor protection system, instrumentation, and controls for engineering safety features, and the emergency electrical power systems. These requirements will be satisfied by analysis or by providing applicable test results. We conclude that the applicant's environmental testing program will provide assurance that the equipment will function under the conditions expected during an accident, and is acceptable.

8.6 Control Room

The design criterion for the control room is to limit the doses received by an operator continuously remaining in the control room for 30 days following a loss-of-coolant accident to five Rem to the whole body and 30 Rem to the thyroid. In applying this design criterion, the applicant will employ the values we have assumed in our accident analyses for the fission product source, the spray removal constant for elemental iodine, and the wind speed. In addition, he will employ our assumption that organic iodides and particulate iodine are not removable from the containment by the proposed engineered safety features. The control room will be equipped with a separate ventilation system that will provide air conditioning and will automatically actuate recirculation of the air upon signal from radiation detectors. The recirculated air will be passed through a filter bank to remove radioiodine. As discussed in Section 3.6, the ventilation system will also reduce halogen concentrations in the event of a postulated accident at Dow.

In the remote event that access to, or habitation of, the control room should be precluded for a relatively long period of time, the capability is provided to permit the plant personnel to shut down the unit and maintain it safely in a hot standby condition for an extended period by means of controls located outside the control room. In addition, the reactor can be brought from hot standby to a cold shutdown condition in approximately one day without access into the control room. These design requirements are consistent with AEC criteria and current practice and thus we conclude that the design of the control room is acceptable.

8.7 Common Mode Failure

The applicant is performing studies of means of preventing common mode failures in the reactor protection system from negating scram action. Studies are also being performed of the consequences of failure to scram in the event of anticipated transients. The applicant has stated (see the ACRS letter attached as Appendix B to this evaluation) that he will maintain flexibility in the engineering design with regard to (1) relief capacity of the primary systems and (2) diverse means of reducing reactivity in order to tolerate the consequences of a failure to scram during anticipated transients. As recommended by the ACRS, we will require that the applicant accelerate the study of means of preventing common mode failures which may negate scram action and, if necessary, will require modifications to the plant to make tolerable the consequences of failure to scram during these transients. Our evaluation of the probability and consequences of these types of events will provide the basis for further review of the proposed design of the systems regarding their ability to terminate or limit the consequences of such events. The applicant will be required to make such changes in the final design as are found necessary as a result of this further review.

9.0 RADIOACTIVE WASTE TREATMENT SYSTEM

9.1 Liquid Radioactive Waste

The liquid waste disposal system, which serves both reactors, is designed to collect, monitor, and process all wastes which are potentially radioactive and to permit the controlled release of radioactive wastes to the Tittabawassee River within the limits specified in 10 CFR Part 20. The major sources of radioactive liquid waste result from the water from the reactor coolant system that is removed and stored during reactor startup and during adjustments in the boric acid concentration of the reactor coolant system, from liquid samples of the reactor coolant system taken for chemical and radioactivity analysis, and from collection of leakage from operating systems. These wastes are vacuum degassed and stored in waste holdup tanks, before they are filtered, demineralized and evaporated. The applicant estimates that a decontamination factor of at least 1,000,000 will result from this processing. The end products, concentrated boric acid and demineralized water, are normally stored for later reuse; however, after sampling they may be released to the river as radioactive liquid waste.

Liquid wastes having a potential for chemical contamination are collected from the radioactive laboratory drains, building sumps, and decontamination shower drains. These wastes are filtered.

They are monitored after filtration and then either discharged to the river or processed by evaporation and demineralization. The applicant estimates that a decontamination factor of at least 10,000 will result from this processing. Following treatment, the water may be either reused or discharged after sampling to the river.

The applicant has calculated the total radioactive content of the various tanks comprising the waste treatment system assuming the tanks contain the wastes resulting from one refueling, four cold startups, two hot startups, and the draining of one steam generator for maintenance. It is assumed that the primary coolant of each unit contains radioactive material equivalent to that which the applicant estimates would result from operation of the reactor with 1% failed fuel. With these assumptions, 313,000 curies of gaseous activity, 41,300 curies of dissolved or suspended liquid activity, and 8,300 curies of tritium would be stored in the waste treatment system. These assumptions are conservative and thus represent an upper limit estimate of the amount of radioactive waste material that would be stored. Approximately 96% of the gaseous and dissolved activity and 86% of the tritium will be contained in the six liquid waste holdup tanks located inside the reactor containments. In the event of a failure of any of these tanks, the radioactive wastes stored in the tanks will be retained in the containment building.

Assuming the operating cycle presented above, the applicant estimates the maximum annual release from the facility would be 6,300 curies of tritium, (2.6% of the 10 CFR Part 20 limits assuming a cooling pond blowdown of 90 cfs), and 0.15 curies of dissolved activity. The largest constituents of the dissolved activity released is anticipated to be 0.015 curies/year of iodine-131 (0.06% of the 10 CFR Part 20 limits), and 0.055 curies/year of cesium-137 (0.003% of the 10 CFR Part 20 limits). In view of the conservatism of the assumptions regarding operating cycle and amount of radioactivity released to the primary coolant from failed fuel, and the small fraction of the 10 CFR Part 20 limits expected, we find these releases to be acceptable.

The effluent from the facility will be continuously monitored. High activity will cause the liquid effluent control valve to close, thus terminating release of liquid effluent to the Tittabawassee River. The nearest municipal water supply which could be affected by releases from the Midland plant is located on Saginaw Bay, 40 to 50 miles from the site. Considering the levels of radioactivity that may be released from the plant, the applicant's proposed environmental monitoring program, and the long transit time to Saginaw Bay which will provide ample time for monitoring the movements of radioactivity and the taking of corrective action, should it be necessary, we conclude that there will be no significant hazard to drinking water supplies as a consequence of normal operation of the Midland plant. To date, operating experience with pressurized water reactor plants indicates that the liquid effluent discharge is only a small fraction of that specified in 10 CFR Part 20.

9.2 Gaseous and Solid Radioactive Waste

The gaseous waste treatment system treats the gases vented from all potentially radioactive systems, and the gases drawn from the liquid waste treatment system by the vacuum degassers. Gases are stored temporarily in the waste gas surge tank where they are monitored. If the radioactive content is high, the gases are compressed and stored in the four waste gas decay tanks until the radioactivity has decayed to a level acceptable for release. The applicant estimates that when operating with the cycle described above, 430,000 curies of gaseous activity will be stored in the gaseous waste system of which 93% would be stored in the four waste gas decay tanks. Under these conditions, the average release rate from the plant would be 42.5 millicuries per second (mCi/sec) of Xe-133, 2 mCi/sec of Kr-85, and 0.6 mCi/sec of Kr-88. The applicant calculates that these releases will not result in concentrations of radionuclides at the site boundary in excess of the 10 CFR Part 20 limits.

At the operating license stage of our review, we will evaluate the onsite meteorological data which will be available and will determine the appropriate diffusion characteristics of the site. We will then develop Technical Specifications that will limit the release rate from the system so that concentrations of radioactive materials in air at the site boundary will be within the limits specified in 10 CFR Part 20.

A monitor is installed in the plant ventilation discharge duct. Radioactive gases will be monitored and discharge of gaseous effluent will be automatically terminated if a concentration in excess of the limit which will be established in the Technical Specifications is reached.

Our evaluation of the consequences of a rupture of one of the waste gas tanks is presented in Section 12.5.

Solid radioactive wastes will be placed in appropriate containers, removed from the site and disposed of at a licensed waste disposal facility.

We conclude that the waste disposal system proposed by the applicant will provide effective control of radioactive wastes generated at the site to assure that routine release concentrations will fall within the Commission's regulations.

When the Midland plant starts operation, we will require that both the liquid and gaseous waste disposal systems be operated in compliance with regulations then in effect.

10.0 AUXILIARY SYSTEMS

We have reviewed the design bases, the mechanical design, and the redundancy requirements (where applicable) for the auxiliary systems proposed for the Midland plant. The systems included in our review were (1) the reactor coolant makeup and purification system, (2) the chemical addition system, (3) the decay heat removal system, (4) the fuel pool cooling system, (5) the shield cooling system, (6) the component cooling system, (7) the service water system, (8) the auxiliary feedwater system, (9) the fuel handling system, (10) the sampling system, (11) the instrument and service air system, (12) the heating, ventilating and air-conditioning systems, (13) the fire protection system, (14) the condensate and feedwater system for the steam generators, and (15) the circulating water system. The design bases for these systems are the same as those for other recently reviewed and approved PWR plants. On the basis of our review of the Midland systems and of other plants using similar systems, we have concluded that these systems will be adequate to perform their intended functions.

The Midland plant uses a cooling pond as the ultimate heat sink for the facility. The pond is approximately 12 feet deep and has a surface area of 880 acres when full. The capacity of the pond is sufficient to provide the cooling water needs of the plant for 100 days without drawing water from the Tittabawassee River. The cooling pond dikes are capable of withstanding the probable maximum flood.

Nevertheless, to provide a source of emergency cooling water in the event that the cooling dikes should fail, an emergency reservoir, having a surface area of 24-acres and located in the northeastern corner of the cooling pond, will be provided by excavation to a depth of six feet below the normal level of the bottom of the cooling pond. This emergency reservoir will have a useable capacity of 70 acre feet. Considering water seepage into the soil, this capacity is sufficient to reject plant decay heat for 30 days without makeup. The applicant will be required to monitor for silting in the emergency pond and, if necessary, to dredge it periodically.

The applicant has performed preliminary calculations to determine the capability of outer slopes of the area fill around plant structures and of the sloping sides of the emergency cooling water reservoir to withstand the design basis earthquake without sliding. These calculations indicate a factor of safety of 1.2 for shallow slip surfaces in the upper part of the fill, and of 1.8 for slip surfaces which intersect pipe lines. The applicant also has calculated a factor of safety of 1.6 for the cut slopes forming the sides of the emergency cooling water reservoir during the design basis earthquake. Further analyses will be made as construction progresses using soil strength values measured under dynamic loading conditions, and soil profiles developed from bore holes in the intake structure area. The applicant has agreed to vary the slope angle and the type of fill as required to attain a factor of safety of not less than 1.1 for shallow slip surfaces not intersecting pipe lines, and not less than 1.5 for deep slip surfaces which intersect pipe lines. We have evaluated the criteria proposed and have determined that they are adequate to assure slope stability during the design basis earthquake.

We have evaluated the adequacy of the capacity of the cooling pond and the emergency reservoir to satisfy plant cooling needs as required. We have determined that an adequate supply of water will be available both to cool the plant during normal operation with low river flow and to reject plant heat following plant shutdown even in the event of failure of the dikes which contain the water in the cooling pond. Therefore, we conclude that the design of the cooling pond and the associated emergency reservoir is acceptable.

11.0 USE OF PROCESS STEAM

Steam from the secondary system will be removed from the main steamline upstream of the turbine and from the moisture separator between the high and low pressure stages of the turbine. This steam will be passed through a system of intermediate shell and tube heat exchangers to generate steam for export to the Dow Chemical Company. The condensate from the intermediate heat exchangers will be returned to the hotwell of the turbine condenser. Feedwater to the intermediate heat exchangers will consist of condensate returned from the Dow Chemical Company and additional makeup water drawn from Lake Huron and then demineralized, as required. The intermediate heat exchanger system will be designed and constructed in accordance with the standards established for those features of the plant associated with the turbine generator system. The intermediate heat exchangers will be designed in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code and the piping will be designed to the ANSI B31.1.0 Piping Code. We consider these to be acceptable.

The steam condensate from the intermediate heat exchangers will be monitored continuously by a gamma radiation monitor. This monitor will provide an alarm when the gross gamma activity in the steam condensate reaches a level of 3×10^{-6} microcuries per cubic centimeter. In addition, batch samples of the steam condensate and the treated Lake Huron water supplied as makeup to the intermediate heat exchangers will be taken. As determined from analysis of these batch samples using sensitive, low level beta counting equipment, the specific activity of the condensate of the steam delivered to Dow will be compared with that of the treated Lake Huron makeup water to determine if leakage of radioactivity from the secondary system of the nuclear plant into the intermediate heat exchanger system has occurred. If detectable leakage occurs, the leaking intermediate exchanger will be isolated. We have evaluated the system proposed and conclude that it provides adequate assurance that the leakage of radioactivity into the process steam will be extremely low and that the radioactivity of the process steam will be essentially at natural background levels and is acceptable.

12.0 ACCIDENT ANALYSIS

12.1 General

In order to assess the safety margins of the plant design, the following plant operating transients were considered by the applicant: (1) uncompensated reactivity changes resulting from fuel depletion and changes in fission product poison concentrations, (2) control rod withdrawal during startup and at power*, (3) dilution of the boron concentration in the coolant, (4) startup of an inactive coolant loop, (5) loss of coolant flow, (6) malpositioning of a control rod, (7) loss of ac electric power, and (8) loss of electrical load. The applicant's criterion for detailed design of the reactor control and protection system is that the system be able automatically to take corrective action to cope with any of these transients.

Preliminary analyses of these transients have been presented in the PSAR. These analyses indicated that no fuel damage occurs. The consequences of these transients will be calculated again when detailed plant design information is available to verify that these

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*To assure that a rod withdrawal accident at startup does not occur while the pressurizer is full, the applicant has agreed to propose a Technical Specification limit requiring verification that the pressurizer level is below a maximum value prior to withdrawing rods.

transients are within the capabilities of the reactor control and protection systems. Based on our evaluation of the information submitted by the applicant and our evaluations of other pressurized water reactor designs at the operating license stage, we conclude that the Midland protection and control system design is such that these transients can be terminated without the core and reactor coolant boundary being damaged, and with no significant offsite radiological consequences.

The applicant and we have evaluated the consequences of potential accidents, including ejection of a control rod, the rupture of a gas decay tank, a steamline break, a steam generator tube rupture, a refueling accident, and a loss-of-coolant accident.

The calculated offsite radiological doses which would result from rupture of a gas decay tank are well within the 10 CFR Part 100 guidelines.

On the basis of our experience with evaluations of the steamline break and the steam generator tube rupture accidents for pressurized water reactor plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations. We will require limits in the Technical Specifications on primary and secondary radioactivity concentrations such that the potential 2-hour doses at the exclusion radius that we calculate for these accidents will be well within 10 CFR Part 100 guidelines. Recently approved Technical Specifications for operating pressurized water reactors include limitations necessary to reduce the calculated consequences of these accidents to this level.

Our evaluations of the refueling accident, the rod ejection accident, and the loss-of-coolant accident are discussed in the following sections.

12.2 Refueling Accident

In our evaluation of the refueling accident we assume that during fuel handling operations, a dropped fuel bundle falls with sufficient force to physically damage all 208 of the fuel rods in the bundle with consequent release of 20% of the noble gases and 10% of the iodines from the damaged rods into the fuel pool water. It is assumed that the accident occurs 24 hours after shutdown and that the dropped fuel bundle has been removed from a region of the reactor core which has been generating twice the average core power. Ninety percent of the iodines released from the damaged fuel rods are assumed to remain in the refueling water. The remaining fission products are assumed to be discharged to the atmosphere by the auxiliary building charcoal filter with an iodine removal efficiency of 90%. We assume the same meteorological conditions as described in Section 12.4 for the loss-of-coolant accident, and assume that all fission products are released within two hours. The resultant calculated doses are 250 Rem to the thyroid and 8 Rem to the whole body at the site boundary. We calculate course of the accident doses at the outer boundary of the low population zone (1 mile) of 90 Rem to the thyroid and 3 Rem to the whole body.

12.3 Rod Ejection Accident

The applicant has analyzed the accidents resulting from the ejection of a single control rod for both beginning-of-life and end-of-life conditions at both full power and zero power. The applicant's analyses indicate that no fuel damage will result from a rod ejection accident at zero power. The worst case analyzed resulted from a rod ejection accident at full power occurring at the beginning of core life. For this case, the applicant has calculated a peak fuel enthalpy of approximately 170 calories per gram, a peak thermal power of 126% of full power, and predicts that 4.1% of the fuel rods will experience departure from nucleate boiling (DNB) conditions.

The applicant's calculations have been performed using a point reactor kinetics model with a limited number of cases analyzed using a space-dependent kinetics model. For the cases analyzed, the point-kinetics model yields a higher peak fuel enthalpy and, therefore, is conservative. However, to assure that no significant fuel damage can occur as a result of a rod ejection accident, we will require that the applicant perform space-dependent kinetics calculations for both beginning-of-life and end-of-life conditions at both ultimate power and at zero power prior to the issuance of an operating license.

We have estimated the potential offsite consequences resulting from this accident assuming that all rods that experience DNB will undergo failure of the fuel rod cladding. Using these assumptions, we calculated 2-hour site boundary doses of 180 Rem to the thyroid and 1 Rem to the whole body and course of the accident doses at the outer boundary of the low population zone of 170 Rem to the thyroid and 1 Rem to the whole body.

12.4 Loss-of-Coolant Design Basis Accident

Although the basis for the design of the emergency core cooling system is to limit fission product release from the fuel, in our conservative calculation of the radiological consequences of the loss-of-coolant accident we have assumed that the accident results in the release of the following percentages of the total core fission product inventory from the core: 100% of the noble gases, 50% of the halogens, and 1% of the solids. In addition, 50% of the halogens released from the core are assumed to plate out on internal surfaces of the containment building or on internal components. We assumed further that (1) 10% of the iodine available for leakage from the containment is in the form of organic iodides, (2) 5% of the iodine available for leakage adheres to particulate matter, and (3) the containment leaks at a constant rate of 0.1% of the containment free volume per day for the first day, and 0.05% per day thereafter, since containment pressure will be reduced. A spray removal coefficient of 2.5 hours^{-1} is used as discussed in Section 7.2 of this evaluation. It is further assumed that the spray does not remove either organic iodides or particulate iodine

We have evaluated the radiological consequences for the following meteorological conditions.

(1) For the first eight hours: Pasquill Type F stability, one meter per second wind speed, nonvarying wind direction, and a volumetric building wake correction factor of one-half used with the cross-sectional area of the containment structure to determine the building wake reduction factor, with a maximum building wake reduction factor of one-third.

(2) From eight hours to twenty-four hours: Pasquill Type F stability, one meter per second wind speed with meander in a 22-1/2° sector.

(3) From one to four days: Pasquill Type F stability and a two meter per second wind speed with a frequency of 60%, and Pasquill Type D stability and a three meter per second wind speed with a frequency of 40%, with meander in the same 22-1/2° sector.

(4) From four days to 30 days: Pasquill Type C, E, and F stability each occurring 33-1/3% of the time with wind speeds of three meters per second, two meters per second, and two meters per second, respectively, and with meander in the same 22-1/2° sector, 33-1/3% of the time.

The breathing rate for a person offsite is assumed to be of 3.47×10^{-4} cubic meters per second for the first eight hours and 1.75×10^{-4} cubic meters per second thereafter. Using these assumptions, we calculate the potential doses at the site boundary for a two-hour period to be 270 Rem to the thyroid and 4 Rem to the whole body. At the low population zone distance of one mile, our calculated potential doses for a 30-day period are 90 Rem to the thyroid and 3 Rem to the whole body.

In calculating the above doses, no credit was given for the effects of the isolation valve seal water system, the penetration pressurization system, or the weld channel pressurization system in reducing containment leakage. Operation of these systems, which interpose a high pressure area between the containment and the outside atmosphere at all points where leakage might be expected, should significantly reduce the leakage rate from the containment, and, thus, would reduce the doses following an accident.

13.0 CONDUCT OF OPERATIONS

13.1 Technical Qualifications

We have reviewed technical qualifications of Consumers Power Company and its contractors to design and construct the Midland facility. The execution of the project is the sole responsibility of Consumers Power Company which has previous nuclear experience through their operation and construction of the Big Rock Point plant and construction of the Palisades plant.

Consumers Power Company has engaged the Babcock & Wilcox Company to design and supply the nuclear steam supply systems, core flooding systems, feedwater controls, reactor control and protection systems, and other related reactor auxiliary systems. Bechtel Corporation and its affiliate, Bechtel Company, have been employed to design and supply the balance of plant equipment, systems, and structures. Bechtel Company will perform the onsite construction of the plant. (In subsequent discussions both Bechtel Company and Bechtel Corporation are referred to as Bechtel). The Babcock & Wilcox Company is currently engaged in the design, construction, and installation of 10 pressurized water nuclear steam supply systems. Bechtel has been actively engaged in design and construction of 23 boiling water reactor and pressurized water reactor nuclear power plants. On the basis of the above considerations, our previous and current evaluations of plants designed and constructed by the contractors, the applicant's experience in operation of the Big Rock Point plant and in construction of both the Big Rock Point plant the Palisades plant, and our contact with project personnel during our review, we conclude that the Consumers Power Company and its contractors are technically qualified to design and construct the Midland plant.

13.2 Operating Organization

Consumers Power Company will review the plant design, equipment, selection, and construction and will participate in acceptance testing as construction progresses. During construction of the facility, the Division of Compliance will monitor the applicant's capabilities to assure that the applicant's expanding commitment to nuclear power does not dilute the technical support organization.

The onsite plant organization closely parallels that proposed for the Palisades plant, with three main groups under the general direction of the plant superintendent. These are maintenance, technical support, and operations groups. We have evaluated the general plant organization and have concluded that it is satisfactory. The applicant proposes to operate the two units with a dual-unit shift composition of one senior licensed operator, three licensed control room operators, and three auxiliary operators per shift. We have informed the applicant that this crew size may not be acceptable, but that prior to the operating license stage of our review, we will review additional information to be provided by the applicant regarding the ability of the proposed shift composition to safely handle both normal and abnormal conditions at the facility. The applicant's minimum qualifications for plant personnel will be in accordance with Section 4 of the Proposed Standard for Selection and Training or Personnel for Nuclear Power Plants prepared by the ANS-3 Committee of the American Nuclear Society (Draft No. 9 or any subsequent approved revision). We consider this to be satisfactory.

Supervisory personnel at the Midland plant will receive training at either the Big Rock Point or the Palisades plants. In addition, a significant number of control room operators assigned to Midland initially will hold operator licenses at either the Big Rock Point of Palisades plants. We consider this proposal to be acceptable; however, we will require more detail concerning the operator training program employed by the applicant for our review at the operating license stage.

13.3 Emergency Planning

The applicant has developed an outline of plans to handle a radiological emergency at the Midland plant. Procedures will be developed which will govern the actions operators and supervisors must take in the event of a radiological emergency. These will include procedures for assuring that the reactor is in a safe condition, that means are available for determining the radiation levels within the plant and at the plant boundary, that methods of controlling access to the plant are provided, that the Dow Chemical Company plant protection supervisor is notified so that the Dow Emergency Action Plan can be implemented, and that Consumers Power Company management personnel and civil authorities are notified if required.

Although only a small portion of the Dow plant lies within the exclusion area of the Midland plant, Dow has agreed to evacuate the entire Dow complex in the event of radiological emergency, if advised to do so by Consumers Power Company. Such an order to evacuate would be initiated by the Consumers Power Company shift supervisor. The Dow Chemical Company has an established plan for emergency evacuation of the Midland Chemical plant. Dow estimates that 90% of the plant personnel can be evacuated from the chemical plant within 20 minutes of receipt of the evacuation signal at the process units, and all can be evacuated within 45 minutes. Most Dow employees work at locations that are from one to three miles from the reactor facility. In an emergency condition, use would be made of available department vehicles to transport personnel to parking lots located approximately 1/2 mile from the center of the Dow site. We have calculated that the dose that might be received by an employee standing one mile from the reactor during a 35-minute period following a design basis loss-of-coolant accident would be 55 Rem to the thyroid. Because most Dow employees are located from one to three miles from the reactors, this calculation represents an overestimate of the dose that might be received by the 90% of the Dow personnel who evacuate within 20 minutes. We have also calculated that the dose that might be received by an employee standing one mile from the reactor for one hour would be 75 Rem to the thyroid. We consider this to be representative of the maximum potential dose which might be received by those Dow employees who must remain on site to shut down Dow facilities. These doses are well below the guideline levels of 10 CFR Part 100. Based on the above, we conclude that the Dow evacuation plans are adequate to assure evacuation of Dow employees in a timely manner.

14.0 QUALITY ASSURANCE

We have reviewed the quality assurance program presented by the applicant for the design, construction, and operation of the Midland plant. The Consumers Power Company will have the final responsibility for the quality assurance program. The applicant has assigned the basic portion of the quality assurance program to Bechtel. The Babcock & Wilcox Company will have day-to-day responsibility for the nuclear steam supply system.

The applicant has assigned responsibility for design, procurement, manufacturing, and shipping phases of the Midland project to the Manager of General Plant Engineering. The Consumers Power Company has assigned the direction and coordination of the quality assurance program from design through construction to the Quality Assurance Engineer (QAE). The QAE reports directly to the Manager of General Plant Engineering and will plan and administer the applicant's quality assurance program, determine the adequacy of the quality assurance plans of Bechtel, B&W, and other contractors or vendors, and take corrective measures to all deviations from the plan. The QAE will be assisted by a field quality assurance engineer. The duties of the QAE field quality assurance engineer are addressed in the PSAR.

Bechtel as the architect-engineer and constructor has prepared six manuals to provide instructions, guidelines, and procedures to assure implementation of the quality assurance program. Significant design aspects orginating within the Bechtel organization will receive at least one internal independent review prior to approval. They are then subject to review and approval by the applicant.

The QAE and the Bechtel Quality Assurance Coordinator will audit the Bechtel quality assurance program to assure that it is being implemented.

The Babcock & Wilcox Company as supplier of the nuclear steam supply system has established a quality assurance program to cover the areas of design, procurement, fabrication, and testing. The B&W Nuclear Power Generation Division (NPGD) Quality Assurance organization administers the quality assurance program and reports directly to the Vice-President in charge of the NPGD. B&W implements the quality assurance program by use of standards and written procedures.

The applicant's QAE, with the assistance of Bechtel, audits the quality assurance and quality control programs of B&W and vendors. B&W will also audit the quality assurance programs of its suppliers as appropriate.

Based on our discussions with the applicant, Bechtel, and B&W, and the information in the application, as amended, we conclude that the Midland plant quality assurance program meets the requirements of the "Nuclear Power Plant Quality Assurance Criteria," Appendix B, 10 CFR 50 and is acceptable.

15.0 RESEARCH AND DEVELOPMENT

A number of areas have been identified for which further analytical, experimental, design development, or testing efforts will be performed to substantiate the adequacy of the pressurized water reactor design. Specific areas requiring attention prior to completion of the design are summarized below.

15.1 Core Stability and Power Distribution Monitoring

This program is required to establish the stability characteristics of the core and demonstrate that the partial length control rod system can control any core instability to assure the desired operation of the plant. The B&W program on xenon oscillations consists of the following analyses:

1. Modal analysis
2. One and two dimensional digital analysis
3. Three dimensional analysis

The results of the modal analysis performed by B&W have been submitted as Topical Report BAW-10010, "Stability Margin for Xenon Oscillations - Modal Analysis." A one-dimensional digital analysis will be used to determine the validity of the modal analysis approach. The results of the one-and two-dimensional digital analyses will be submitted as a topical report shortly. The three-dimensional digital analysis results will be submitted for our review later this year.

The entire program is scheduled for completion well before the scheduled startup of the Midland plant.

Information is needed to demonstrate that sufficient information can be derived from external detectors alone to determine the power distributior after the reactor has been operated. The flux distribution will be perturbed because the axial burnup is not uniform, and because of the effects of fuel or control rod replacement or eerrors in fuel element position or enrichment. In addition, little experience exists with operation of large power reactors to ascertain how frequently out-of-core detectors should be recalibrated. If the planned research and development program does not produce convincing evidence that the out-of-core detection system is sufficient, we will require that a minimum number of in-core detectors, properly positioned throughout the core, be operable at all times when the reactor is operating at power.

15.2 Fuel Rod Clad Failure

The Babcock & Wilcox Company has initiated a study of fuel clad failure mechanisms associated with a loss-of-coolant accident that includes an evaluation of existing data and scoping tests to obtain data on potential fuel clad failure mechanisms. These tests include studies of eutectic formation, brittle failure, and clad swelling.

An analytical study of fuel clad failure is in the planning stage. This program will consist of an evaluation of the axial and radial temperature distributions throughout the core. The change in flow channel resistance to flow was calculated and incorporated into the channel analysis. This program is designed on the basis that the major unknown is the amount and location of flow blockage that could result from clad deformation in a loss-of-coolant accident.

Multi-pin tests will provide data to determine the possible interaction between pins undergoing a temperature excursion. These data, together with data from the FLECHT program (Full Length Emergency Cooling Heat Transfer Test), scheduled for completion in 1970, will provide further information on the capability of the emergency core cooling system to function as designed. These data will be used in conjunction with improved multi-node analytical techniques to verify the performance of the emergency core cooling system. In addition to this research and development program, we will require the applicant to analyze the consequences of partial melting and subsequent disintegration of a portion of a fuel assembly at the operating license stage of our review, as recommended by the Advisory Committee on Reactor Safeguards.

15.3 Internals Vent Valves

An experimental program has been performed by the Babcock & Wilcox Company to verify the performance of the internals vent valve assemblies. The program included a hydrostatic test, valve disc closing test, tests to verify the pressure differences to open the valve discs and maintain the valve disc in a maximum open position, handling test, a vibration test, and a test of prototype valves in a 1/6 scale model of the reactor vessel and internals. This test program has been completed. We are presently evaluating the report of the program.

15.4 Once-Through Steam Generator

The Babcock & Wilcox Company has conducted tests on 7-tube, 19-tube, and 37-tube mockups of the once-through steam generator to investigate heat transfer, heat capacity, control and dynamic response, structural integrity, vibration, feedwater heating, tube leakage propagation, and simulated steamline failure. The program has been completed. We are reviewing a report of the program at the present time to determine if it provides sufficient justification to permit us to accept the applicant's conclusion that the tests substantiate the acceptability of the design.

15.5 Reagent Spray System

The Babcock & Wilcox Company has performed tests on sodium thiosulfate solution stability under storage and accident conditions. We are presently evaluating the report of the results of these tests. Material compatibility studies have been conducted on the types of metals used in the primary system and in the recirculation portion of the emergency core cooling system. Testing to date has included stressed specimens and tests are planned of welded samples. When the material compatibility [illegible]g, including tests of welded specimens, has been completed, we will complete our evaluation of the acceptability of the sodium thiosulfate solution for the spray system. The applicant has agreed to reserve space for installation of charcoal filters should the research and development program fail to meet its objectives.

15.6 Process Steam Monitoring

The applicant will conduct a research and development program to verify the required sensitivity of the proposed gross gamma monitor on the steamline carrying export steam to Dow. These tests will be performed at the Consumers Power Company Palisades plant. All tests will be completed prior to the submittal of the Final Safety Analysis Report.

15.7 Control Rod Drive Test

The Babcock & Wilcox Company control rod drive test program to develop the roller-nut type drive has been completed. We are presently evaluating the report of the results. Several areas have been identified to B&W where more details of the tests results should be addressed.

15.8 Self-Powered Detector Tests

The B&W research and development program for self-powered detectors has been completed (longevity testing is continuing) and reported to us. The testing of the self-powered detectors has indicated that this system is capable of measuring neutron flux in a pressurized water reactor environment with a relative accuracy of ± 5 percent over a three year time span. This device has an inherently large time constant and is not used in any direct safety actions. As indicated in Section 15.1 of this evaluation, if out-of-core detectors are not capable of detecting core instability, at the operating license review stage we will establish the minimum number of incore instruments that must be operable when the reactor is operated at rated power.

15.9 Core Thermal & Hydraulic Design

B&W is conducting a research and development program for heat transfer and fluid flow investigations. The requirements of the experimental programs are developed from the thermal and hydraulic core design limits set forth in Section 3 of the PSAR. We are presently reviewing a report, on these tests. We will continue to review these matters to assure that sufficient safety margin is available to prevent events which could cause departure from nucleate boiling and subsequent fuel failures.

15.10 Blowdown Forces on Core Internals

The stresses and deflection of the reactor internals have been analyzed by B&W. The results of this analysis have been reported and are currently being reviewed.

15.11 Conclusion

Based on our review of the research and development programs proposed, we conclude that these programs are timely, are reasonably designed to accomplish their respective development objectives, will provide adequate information on which to base analyses of the design and performance, and should lead to acceptable designs for the systems involved.

16.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS) has completed its review of the application for construction permits for the Midland plant Units 1 and 2. Copies of the ACRS letters dated June 18, 1970, and September 23, 1970, are attached as Appendix B. The letters contain several recommendations and note several items to be resolved by the applicant and the staff during construction. These matters are discussed in this safety evaluation in the sections indicated: (1) onsite meteorological program (Section 3.2), (2) limit on the chlorine concentration in the control room following an accidental release at the Dow plant (Section 3.6), (3) means of prompt detection of fuel failure (Section 4.0), (4) review of criteria and procedures used for the installation of protection and emergency power systems together with appropriate procedures to maintain the physical and electrical independence of the redundant portions of these systems (Section 8.4), (5) addition of a high containment pressure reactor trip signal (Section 8.1), (6) review of the results of improved analytical techniques used to analyze the loss-of-coolant accident and the capability of emergency core cooling system (Section 7.1 and Section 15.2), (7) review of procedures for implementation of the criteria established regarding export of process steam to the Dow Chemical Company (Section 11.0), (8) development of systems to control the concentration of hydrogen in the containment which might accumulate in the unlikely event of a major accident (Section 7.4), (9) review of the applicant's study of means of preventing common mode failures and of the consequences of failure to scram during anticipated transients (Section 8.5), (10) review of analysis of the consequences of melting and subsequent disintegration of a portion of a fuel element (Section 15.2), and (11) information on items identified in previous ACRS reports on other reactors (Section 15.0).

The ACRS concluded in its September 23, 1970 letter, that these items "...can be resolved during construction and if due consideration is given to these items and to the items referred to in its June 18, 1970 report, the nuclear units proposed for the Midland plant can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public."

17.0 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's organization are citizens of the United States. We find nothing in the application to suggest that the applicant is owned, controlled, or dominated 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 Paragraph 50.33(j) of 10 CFR Part 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 for military purposes is involved. For these reasons, and in the absence of any information to the contrary, we conclude that the activities to be performed will not be inimical to the common defense and security.

18.0 FINANCIAL QUALIFICATIONS

Based upon the evaluation of the financial information presented in the application, Amendment No. 13 and in the 1969 and previous Annual Reports of the company, it is the staff's opinion that the Consumers Power Company, a Michigan corporation, is financially qualified to design and construct the nuclear generating station to be known as the Midland Plant Unit Nos. 1 and 2.

The estimated cost of construction for both units of the nuclear facility, including costs for the first core fuel for each unit, is $394,827,000 of which $346,640,000 is for the nuclear production plant, $3,145,000 is for associated plant and $45,042,000 is for the nuclear fuel for the initial cores. We have determined that the estimated costs of production plant construction are reasonable and the fuel requirements for the first core of each unit are reasonable.

The applicant will finance the total costs to construct the Midland plant ($394.8 million) as an integral part of its normal construction program, using funds internally generated (cash on hand, undistributed earnings and depreciation and other accruals) and from the sale of securities (debt, equity and short-term notes) when and as required, in the same general manner as it finances other plant additions.

An analysis of the applicant's financial statements over the past six years (1964-1969) indicates a strong financial position, sound financing, adequate resources and a high level of earnings. This analysis, together with the reasonable assumption that such earnings will continue, the applicant's excellent credit and bond ratings and its proven ability to borrow on a short-term basis, supports the conclusion that the applicant will be able to obtain the funds from the sources indicated. A detailed evaluation is attached as Appendix H.

19.0 CONCLUSIONS

Based on the proposed design of the Midland Plant Unit Nos. 1 and 2 of the Consumers Power Company; on the criteria, principles, and design arrangements for systems and components thus far described, including all of the important safety items; on the calculated potential consequences of routine and accidental release of radioactive materials to the environs; on the scope of the development program which will be conducted; and on the technical competence of the applicant and the principal contractors; we have concluded that, in accordance with the provisions of Paragraph 50.35(a), 10 CFR Part 50, and Paragraph 2.104(b), 10 CFR Part 2:

1. The applicant has described the proposed design of the facilities, including the principal architectural and engineering criteria for the design, and has identified the major features or components for the protection of the health and safety of the public;
2. Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration will be supplied in the Final Safety Analysis Report;

IMAGE EVALUATION TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

3. Safety features or components which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety question associated with such features or components;
4. On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion or construction of the proposed facility, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public;
5. The applicant is technically qualified to design and construct the proposed facility;
6. The applicant is financially qualified to design and construct the proposed facility; and
7. The issuance of a permit for the construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

APPENDIX A

CHRONOLOGY

1.	January 13, 1969	Consumers Power Company formally filed Application for Licenses for the Midland Plant, Units 1 and 2.
2.	January 22, 1969	ACRS Subcommittee meeting at site.
3.	February 3, 1969	Submittal of Amendment No. 1. Results of the foundation investigation phase of the environmental study at the proposed Midland Plant together with a report "Foundation Investigation and Preliminary Exploration for Borrow Materials."
4.	February 4, 1969	ACRS Subcommittee Meeting to discuss Midland Plant site.
5.	February 5, 1969	Meeting with applicant to discuss meteorological studies.
6.	February 6, 1969	ACRS meeting with applicant to discuss Midland Plant site.
7.	March 21, 1969	Meeting with applicant to discuss Midland Plant site.
8.	March 28, 1969	Letter to applicant concerning acceptability of Midland Plant site.
9.	May 27, 1969	Letter to applicant transmitting comments of the U. S. Fish and Wildlife Service.
10.	May 28, 1969	Submittal of Amendment No. 2. Revised and additional pages and figures for incorporation in the PSAR, incorporating several design changes in response to AEC-DRL letter of March 28, 1969.

11.	July 15, 1969	Meeting with applicant to discuss the general design of the Midland Plant.
12.	July 24, 1969	Meeting with applicant to discuss containment structural design and site geology.
13.	August 13, 1969	Submittal of Amendment 3. Supplement to the Dames and Moore Foundation Investigation Report submitted by Amendment No. 1 to the PSAR.
14.	September 26, 1969	Letter to applicant requesting additional information on site, reactor design, reactor coolant system design, structural design, engineered safety features and other miscellaneous items.
15.	October 2, 1969	Submittal of Amendment No. 4. Revised Section 6.2 and Figure 6-4 of the PSAR, relating to the reactor building spray system, and Appendix 1B of the PSAR, which described the Quality Assurance Program.
16.	October 30, 1969	Meeting with applicant to discuss Quality Assurance Program.
17.	November 7, 1969	Submittal of Amendment No. 5. Amended and additional pages for substitution in PSAR and responses to AEC regulatory staff's request for additional information of September 26.
18.	December 5, 1969	Meeting with applicant to discuss Amendment No. 5.
19.	December 16, 1969	Meeting with applicant to discuss COPATTA Code.
20.	December 29, 1969	Submittal of Amendment No. 6. Revises and supplements information in PSAR and portions of information submitted by Amendment No. 5.
21.	January 8, 1970	Request to applicant for additional information on reactor site, design, coolant system design and miscellaneous other topics.
22.	January 20, 1970	Meeting with applicant to discuss Quality Assurance, meteorology, emergency power and tornado design.

23.	January 30, 1970	Submittal of Amendment No. 7. Revised pages, amending the responses given in Amendment 5 and 6 and response to the AEC regulatory staff's request for additional information dated January 8, 1970.
24.	February 10, 1970	Submittal of Amendment No. 8. Revises and supplements the PSAR, the applicant's responses contained in Amendments 5, 6 and 7 and the applicant's Quality Assurance Program.
25.	February 26, 1970	Meeting with applicant to discuss subsidence, flooding, and slope stability.
26.	February 26, 1970	Request to applicant for additional information on flooding.
27.	March 12, 1970	Meeting with applicant to discuss request for information regarding the maximum probable flood and to identify the additional information required in order to complete evaluation on subsidence.
28.	March 19, 1970	Meeting with applicant to discuss seismic design.
29.	March 24, 1970	ACRS Subcommittee meeting.
30.	April 1, 1970	Meeting with applicant to discuss subsidence.
31.	March 30, 1970	Submittal of Amendment No. 9. Response to AEC regulatory staff's request for additional information of 2/26/70 on hydrology and slope stability and other items and additional information on control room design.
32.	April 24, 1970	ACRS Subcommittee Meeting.
33.	April 28, 1970	Submittal of Amendment No. 10. Revised pages, report by General Analytics, and logs covering salt and brine well operations of Dow Chemical Company.
34.	April 30, 1970	Meeting with applicant to discuss items raised by ACRS subcommittee.

35.	May 1, 1970	Submittal of Amendment No. 11. Revised pages to PSAR.
36.	May 6, 1970	Meeting with applicant to discuss Dow usage of process steam.
37.	May 25, 1970	Meeting with applicant to discuss Dow usage of process steam.
38.	May 28, 1970	Submittal of Amendment No. 12. Revised and additional pages on reactor vessel integrity, analysis of hazardous chlorine release and ground surface subsidence.
39.	May 28, 1970	Submittal of request for an exemption to requirements of 10 CFR 50.10(b).
40.	May 28, 1970	Submittal of Amendment No. 13. Updated corporate and financial information of the "Application for Licenses".
41.	June 10, 1970	ACRS Subcommittee meeting.
42.	June 18, 1970	ACRS issues letter regarding the Midland plant.
43.	July 23, 1970	Meeting with applicant to discuss containment design pressure.
44.	July 30, 1970	Issuance of exemption to requirements of 10 CFR 50.10(b).
45.	July 31, 1970	Submittal of Amendment No. 14. Modification of reactor building design to reflect changes made in the design pressure of the reactor buildings to meet current design parameters.
46.	September 10, 1970	Meeting with applicant to discuss use of tertiary heat exchanger.
47.	September 4, 1970	Submittal of Amendment No. 16. Information on tertiary heat exchanger system.

48.	September 11, 1970	Submittal of Amendment No. 17. Information on tertiary heat exchanger system.
49.	September 14, 1970	ACRS Subcommittee meeting.
50.	September 15, 1970	Submittal of Amendment No. 18. Information on tertiary heat exchanger system.
51.	September 18, 1970	ACRS meeting.
52.	September 23, 1970	ACRS issues letter regarding the Midland plant.

APPENDIX B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

June 18, 1970

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

Subject: REPORT ON MIDLAND PLANT UNITS 1 & 2

Dear Dr. Seaborg:

During its 122nd meeting, June 11-13, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by the Consumers Power Company for a permit to construct the Midland Plant Units 1 and 2. During this review, the project also was considered at Subcommittee meetings held on January 22, 1969, at the plant site, on April 24, 1970, at Chicago, Illinois, on February 4, 1969, March 24, 1970, and June 10, 1970, at Washington, D. C. and at the ACRS meetings of February 6, 1969, April 9, and May 8, 1970, in Washington, D. C. In the course of these meetings, the Committee had the benefit of discussions with representatives and consultants of the Consumers Power Company, Babcock and Wilcox Company, Bechtel Corporation, Dow Chemical Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Midland Plant site is on the south bank of the Tittabawassee River adjacent to the southern city limits of Midland, Michigan. The main industrial complex of the Dow Chemical Company lies within the city limits directly across the river from the site and provides an area of controlled access about two miles wide between the reactor site and the Midland business and residential districts. The exclusion area of the plant site has a radius of 0.31 miles and includes a small segment of the Dow plant; no Dow employees are permanently assigned in this segment, and the applicant has the right to remove any persons from this segment if conditions warrant. The low population zone has a radius of 1.0 miles and contains 38 permanent residents and about 2,000 industrial workers, mainly employees of Dow Chemical Company. The number of permanent residents within five miles of the plant site was estimated to be 41,000 in 1968, mainly in the city of Midland and its environs.

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The applicant has established criteria for, and has begun the formulation of a comprehensive emergency evacuation plan. This plan is being coordinated with the well-established plan of the Dow Chemical Company for emergency evacuation of the Midland chemical plant and portions of the City of Midland in case of major emergencies at the chemical plant. Close coordination with appropriate municipal and state authorities is also being established.

The Midland units will each include a two-loop pressurized water reactor designed for initial core power levels up to 2452 MWt. The nuclear steam supply systems and the emergency core cooling systems of these units are essentially identical with those for the previously reviewed Oconee Units 1, 2 and 3 and Rancho Seco Unit 1 (ACRS reports of July 11, 1967 and July 19, 1968, respectively). The combined electrical output of the two units will be 1300 MW. In addition, 4,050,000 lbs per hour of secondary steam will be exported to the adjacent Dow plant to supply thermal energy for chemical processing operations.

The prestressed, post-tensioned concrete reactor containment buildings are similar to those approved for the Oconee Units 1, 2 and 3. The design will include penetrations, which can be pressurized, and isolation valve seal water systems to reduce leakage. Channels will be welded over the seam welds of the containment liner plates to permit leak testing of the seam welds.

Cooling water for the Midland reactors is supplied from a diked pond with a capacity of 12,600 acre-feet. Make-up water is taken from the Tittabawassee River. The cooling water supply is sufficient for 100 days of full power operation without make-up during periods of low river flow. In the unlikely event of a gross leak through the dikes of the cooling pond, a supplemental source of water will be available. The supplemental source is provided within the main pond by excavating a 24 acre area to a depth of six feet below the bottom of the main pond. This source can supply shut-down cooling capability for 30 days without make-up.

The applicant will conduct an on-site meteorological monitoring program to verify the applicability of the meteorological models used for accident evaluation and routine release limits as well as to determine any meteorological effect of the cooling pond. This program should be completed during construction.

Midland is the first dual purpose reactor plant to be licensed for construction. The export steam originates from the secondary side of the steam generators and may contain traces of radioactive leakage from the primary system. The demineralized condensate from 60 to 75 percent of the export steam is returned by Dow to the feed water supply of the reactor plant. The condensate from the remaining steam is either chemically contaminated or cannot practically be returned to the nuclear plant. It is collected in the Dow waste treatment system for dilution and processing with other streams before eventual discharge to the river. Thus, the unreturned portion of the condensate represents an effluent from the reactor plant to which the requirements of 10 CFR Part 20 must apply.

This matter may be considered in two parts: (1) the steps taken by the applicant to ensure that any radioactivity in the export steam is within the limits set by 10 CFR Part 20 and as low as practicable and (2) the measures taken by the Dow Chemical Company to ensure that the export steam can be used in chemical operations without product contamination and that the unreturned steam condensate is properly managed for safe disposal. In connection with item (1), the applicant proposes to monitor and control radioactivity in the export steam. A representative, continuous sample of the export steam will be condensed for monitoring and laboratory analysis. The gamma activity of this flowing sample will be continuously monitored by on-line analyzers and an alarm actuated if the activity exceeds an appropriate limiting value. The alarm will serve to indicate any change in the integrity of the steam generators or fuel cladding. Samples of this condensate stream will be analyzed at appropriate intervals by sensitive low-level beta counting for determination of gross beta activity and concentration of selected radionuclides. The applicant agrees to limit, by maintaining high integrity of the steam generators and fuel cladding, the yearly average gross beta activity in the export steam to one-tenth or less of the limits specified by 10 CFR Part 20 for the selected radionuclides. The yearly average will include any periods of short duration when the concentrations may approach but not exceed the 10 CFR Part 20 limits. The applicant states that in his judgement it is practical to operate the plant within these limits. If these limits are exceeded, corrective measures will be taken in the plant or the delivery of export steam to Dow will be terminated. He also agrees to demonstrate the analytical equipment and procedures in development programs to be carried forward and completed during construction of the Midland Plant. In connection with item (2), Dow has stated that they will apply for a 10 CFR Part 30 Materials License to receive, possess, and use the export (secondary) steam as a source of thermal and mechanical energy. No export steam or condensate will be intentionally introduced into any product. Isolation of the export steam from contact with products will be accomplished by the use of heat exchange devices which will provide suitable physical barriers. Programs will be established to provide for detection of leaks in the heat exchange devices by analyses, monitors, and other means; for repair of leaks when detected; and for appropriate administrative control of the programs.

Dow has stated that accumulation of radioactivity from the export steam and release of radioactive materials in the effluent will be in accordance with 10 CFR Part 20. The unreturned condensate will represent less than 10% of the total liquid effluent disposed of through the Dow waste treatment plant and the annual average concentration in the total effluent is expected to be less than 1% of the 10 CFR Part 20 limits.

The Committee believes that the criteria proposed by the applicant and Dow for the control of radioactivity in the export steam are necessary and adequate. The detailed procedures for implementation should be developed during construction in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

To minimize the likelihood of subsidence at the site, the applicant and Dow have agreed to prohibit future salt mining operations within one-half mile from the center of the reactor plant. No new wells will be drilled within this distance and all existing wells will be abandoned and plugged. The Committee believes these arrangements are satisfactory.

A large volume of liquid chlorine is maintained in a refrigerated storage vessel about one mile from the Midland plant control room. The applicant is continuing his study of the consequences of a major accidental release of chlorine from this vessel. He has included in his criteria for the design of the control room the objective of finding a practical method of maintaining the concentration of chlorine in the control room atmosphere below the eight hour threshold limiting value (TLV) of 1 ppm for the most serious conceivable chlorine accident. The Committee believes that adequate air purification facilities should be provided in the control room ventilation system to reduce chlorine concentration to the eight hour TLV of 1 ppm so that operators can work without respiratory equipment during an extended chlorine emergency. This matter should be resolved during construction in a manner satisfactory to the Regulatory Staff.

The reactor vessel cavity will be designed to withstand mechanical forces and pressure transients comparable to those considered in the design of the Zion and Indian Point-3 plants.

The applicant has stated that he will provide additional evidence obtained by improved multi-node analytical techniques to assure that the emergency core cooling system is capable of limiting core temperatures to the limits established at present. He will also make appropriate plant changes if the further analysis demonstrates that such changes are required. This matter should be resolved during construction in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The safety injection system for the Midland plant is actuated by either low reactor pressure or high containment pressure signals. However, of these two, the reactor is tripped only by the low reactor pressure signal. The Committee believes that provision also should be made to trip the reactor by the high containment pressure signal.

The applicant plans to develop more detailed criteria for the installation of protection and emergency power systems together with appropriate procedures to maintain the physical and electrical independence of the redundant portions of these systems. The Committee believes that these criteria and procedures should be reviewed and approved by the Staff prior to actual installation.

The applicant considers the possibility of melting and subsequent disintegration of a portion of a fuel assembly because of flow starvation, gross enrichment error, or from other causes to be remote. However, the resulting effects in terms of local high temperature or pressure and possible initiation of failure in adjacent fuel elements are not well known. Appropriate studies should be made to show that such an incident will not lead to unacceptable conditions.

The Committee believes that consideration should be given to the utilization of instrumentation for prompt detection of gross failure of a fuel element.

The Committee has commented in previous reports on the development of systems to control the buildup of hydrogen in the containment which might follow in the unlikely event of a major accident. The applicant proposes to make use of a technique of purging through filters after a suitable time delay subsequent to the accident. However, the Committee recommends that the primary protection in this regard should utilize a hydrogen control method which keeps the hydrogen concentration within safe limits by means other than purging. The capability for purging should also be provided. The hydrogen control system and provisions for containment atmosphere mixing and sampling should have redundancy and instrumentation suitable for an engineered safety feature. The Committee wishes to be kept informed of the resolution of this matter.

The Committee recommends that the applicant accelerate the study of 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 stated that the engineering design would maintain flexibility with regard to relief capacity of the primary system and to a diverse means of reducing reactivity. This matter should be resolved in a manner satisfactory to the Regulatory Staff during construction. The Committee wishes to be kept informed.

Other problems related to large water reactors have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports. The Committee believes that resolution of these items should apply equally to the Midland Plant Units 1 & 2.

The Committee believes that the above items can be resolved during construction and that, if due consideration is given to these items, the nuclear units proposed for the Midland Plant can be constructed with reasonable assurance that they can be operated without undue risk to the health fety of the public.

Sincerely yours,

Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References

1) Amendments 1 - 12 to License Application

APPENDIX B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

September 23, 1970

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

Subject: SUPPLEMENTAL REPORT ON MIDLAND PLANT UNITS 1 AND 2

Dear Dr. Seaborg:

At its 125th meeting, September 17-19, 1970, the Advisory Committee on Reactor Safeguards completed its review of amendments to the application by the Consumers Power Company to construct the Midland Plant Units 1 and 2. This project was the subject of a report to you dated June 18, 1970. The review was reopened in consideration of additional submittals by the applicant proposing an increase in the design pressure of the containment structure and the addition of a system of reboilers for the generation of steam to be exported to the Dow Chemical Company. These changes were considered at a Subcommittee meeting held in Washington, D. C. on September 14, 1970. The Committee had the benefit of discussion with representatives and consultants of the Consumers Power Company, Babcock and Wilcox Company, Bechtel Corporation, Dow Chemical Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The applicant has revised downward his estimate of the free volume and internal surface area of the containment structure and has revised upward to 60 psig the calculated peak containment pressure reached in the unlikely event of a loss of coolant accident. The containment design pressure has been raised to 67 psig to provide a suitable margin above the peak accident pressure, and an increased number of prestressing tendons will be provided in the containment structure to accommodate the increased pressure. No changes in the structural design criteria are proposed. The Committee believes these changes are satisfactory.

In the earlier design the export steam was taken from the secondary side of the main steam generators and might contain traces of radioactive leakage from the primary system. The applicant now proposes to use this steam in a system of shell and tube reboilers to generate tertiary steam for export to the Dow Chemical Company. Secondary steam condensate from the reboilers is returned to the turbine condenser hot well while feed water for the tertiary side of the reboilers is supplied by condensate from the tertiary steam which is supplemented as required by

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demineralized water from Lake Huron. Blowdown from the reboilers is normally routed to the Dow waste treatment system for disposal to the river but may be sent to the radwaste system of the nuclear plant if secondary to tertiary leakage is detected.

The applicant proposes to install monitoring and analytical facilities to determine the levels of radioactivity in the export steam as described in the June 18, 1970, letter; these include an on-line analyzer for gamma activity and sensitive low level beta counting equipment for analysis of samples of the condensed steam. The applicant expects that the tertiary steam delivered to Dow will contain no more radioactivity than the treated make-up water from Lake Huron. Recycling tertiary steam condensate may result in some slight concentration of naturally occurring radioactivity in the reboiler system but is not expected to effect the validity of the comparison between steam and make-up water radioactivity as a sensitive indication of leakage in the reboilers. If detectable leakage occurs, corrective action will be taken in the plant or delivery of export steam will be terminated.

The applicant agrees to demonstrate the analytical equipment and procedures in development programs to be carried forward during construction of the Midland Plant.

The Committee believes that the proposed system of reboilers will provide substantial additional assurance that leakage of primary system radioactivity into the export steam can be maintained at an extremely low and insignificant level and that the export steam can be maintained essentially at natural background levels. The detailed procedures for monitoring and control of the reboiler system should be developed during construction in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The Committee believes that the above items can be resolved during construction and if due consideration is given to these items and to the items referred to in its June 18, 1970 report, the nuclear units proposed for the Midland Plant can be constructed with reasonable assurance that they can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References

1) Amendments 14-18 to the License Application

APPENDIX C

Comments on

Midland Plant Units 1 and 2
Consumers Power Company
Preliminary Safety Analysis Report
Volumes I and II dated October 30, 1968

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
February 3, 1969

The location of the site in the east central part of Michigan in flat terrain where elevations range between 600 and 625 feet above mean sea level, would indicate that atmospheric flow is largely governed by the large-scale, continental pressure patterns. Thus, in winter and spring when frequent storm tracks pass through the area, the ventilation rate would be high and atmospheric diffusion relatively good. From the Climatic Atlas of the United States [1] this region of Michigan shows an average annual wind speed of about 11 mph, with a maximum of 13 mph in March and a minimum of 8 mph in August.

The immediate approach to the plant from the south, west, and east is over rural, often marsh-like terrain uninterrupted by large buildings. The approach from the north includes the surface roughness and heat source effects of the city of Midland and the Dow Chemical complex. However, this effect would be largely dissipated by the time the flow reached the southern site boundary. An on-site measurement of pertinent meteorological parameters such as the standard deviation of the horizontal wind (σ_θ) and the wind speed ($\bar{u}$) would inherently include the distant upwind turbulent effects provided the effect of the reactor building complex wake could be avoided.

The only near on-site wind data available is a 5-year record from two Dow Chemical wind stations about 1-1/2 miles to the northwest and the Saginaw Tri-City Airport climatological record about 8 miles to the southeast. The Dow station shows an average annual wind speed of 6.8 mph while the Saginaw station shows a value of 10.3 mph. The frequency of winds of 3 mph or less (including calms) is 14% for Dow and 8% for Saginaw. It is difficult to explain the rather low wind speeds at Dow, especially since the data were taken atop a 60-ft telephone pole whereas the Saginaw data were taken at a height of 20 feet. The Saginaw data more nearly agree with the climatological wind data for the region.

The average monthly gustiness data for Dow (Table 2A-11) indicates that in September 1966 the atmospheric diffusion rate was less than Pasquill Type E at a speed of 2 m/s for about 50% of the time during the sunrise hours. Since no joint frequency distribution data between gustiness and wind speed are given, it is not possible to quantitatively asses the probability of specific diffusion rates.

Beside the reservations the applicant has with regard to the use of the Dow data (see p. 2A-32), we have the following reasons for questioning the validity of the Dow data in assessing the atmospheric diffusion from a ground source at the Midland nuclear site: 1) wind speeds which seem unusually low when compared to the climatological averages of the region, 2) the difficulty in being able to classify "gustiness" by the range of azimuth wind direction under low wind speeds (10% calm or 1 mph during September 1966 and 1967), and 3) since a ground source is postulated, the Dow wind data at a 60-ft height above the ground may not be appropriate.

The basis for the applicant's 2-hour diffusion model is the method by which routine hourly weather data (Saginaw) is used to obtain Pasquill diffusion categories. Nine months of data were chosen on the basis of being the "worst" diffusion months as judged by the Dow "gustiness" data. Each hour of the 270 days of Saginaw data were then categorized as to Pasquill Type and wind speed. It should be pointed out that this method is an approximate one which is used when more precise categorization, as with σ_θ, is not possible. The method, by definition, limits Pasquill Types E and F to nighttime hours and conversely limits Types A, B and C to daytime hours. The applicant selected from each of the 270 days the "worst" consecutive two-hour period and then <u>averaged</u> the data over the whole sample to produce the statistics for the model. Thus, the first hour of the period contained 219, 32, and 19 hours respectively for categories F, E, and D and the second hour contained 192, 46, and 32 hours, respectively. The average wind speed for all the F cases was about 2 m/s.

In summary, since the Saginaw data shows that over a period of nine non-consecutive, "worst", months the frequency of moderate to strong inversions (Type F) existed about 20% of the time at a speed of 2 m/s, it would seem reasonably conservative to assume for the 2-hour postulated release of radioactivity, a diffusion rate equivalent to Type F and 1 m/sec. The resulting relative concentration at a distance of 1170 m would be 5×10^{-4} sec m^{-3} as compared to the applicant's value of 1.75×10^{-4}.

Reference

[1] U. S. Dept. of Commerce (ESSA), "Climatic Atlas of the United States", June 1968, 80 pp.

APPENDIX C

Comments on

Midland Plant Units 1 and 2
Consumers Power Company
Preliminary Safety Analysis Report
Amendment No. 2 dated June 5, 1969

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
July 28, 1969

The additional meteorological data presented in Amendment 2 is the analysis of 5 years of routine hourly weather data from Saginaw to obtain a frequency distribution of Pasquill diffusion categories. To quote from our previous comments (2/3/69), "it should be pointed out that this method is an approximate one which is used when more precise categorization, as with σ_θ, is not possible. The method, by definition, limits Pasquill Types E and F to nighttime hours and conversely limits Types A, B and C to daytime hours". A number of recent final safety analysis reports where it was possible to compare the two categorization techniques (hourly weather versus σ_θ data) show serious discrepancies. For example, an analysis of σ_θ data from a Great Lakes reactor site (Docket 50-266) compared to an analysis of routine hourly weather data from the nearest Weather Bureau Station shows a 36 percent frequency of Type D (neutral) for the σ_θ approach compared to 65 percent for the hourly weather approach. Furthermore, the σ_θ approach showed a 51 percent frequency in the three stable categories while the other approach showed 22 percent. From this we would conclude that the unusually high neutral categorization is arbitrary and erroneous and tends to underestimate the stable category frequencies. Since the same high frequency for Type D is shown in the Saginaw hourly data (64 percent, Table 2A-14b) we feel the 11 percent frequency for Type E and 12 percent for Type F would be underestimated by at least a factor of 2 if the σ_θ approach had been used.

The only σ_θ data available for the site are those from the Dow meteorological installations and are summarized in the original report. The summarization, however, is by means of gross averages with no frequency distribution between σ_θ, wind speed and wind direction.

In summary, we see no reason to change our conclusion expressed in the comments of 2/3/69 stating that it would seem reasonably conservative to assume for the postulated 2-hr ground release a diffusion rate equivalent to Pasquill Type F and 1 m/sec. For the new site boundary of 400 m this would result in a relative concentration of 1×10^{-3} s m^{-3}, allowing a factor of 3 for the diffusion effect of the building. This compares to the applicant's value of 2×10^{-4} s m^{-3}. Part of the difference is due to the use of a building diffusion factor of 5.9 by the applicant as compared to our factor of 3.

APPENDIX C

Comments on

Midland Plant Units 1 and 2
Consumers Power Company
Preliminary Safety Analysis Report
Amendment No. 5, dated November 7, 1969

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
January 5, 1970

Except for the revised site boundary of 500 m, no new meteorological data are presented in Amendment No. 5 that would change our conclusions as stated in comments dated February 3, 1969 and July 28, 1969.

For the 2-hour release we conclude that the Type F inversion condition, a 1 m/sec wind speed, and a factor of 3 for building effect is a reasonably conservative assumption.

For the 24-hour release, the applicant's analysis from a very limited amount of data shows that for 18 selected "worse" days, the number of hours of inversion Type F varied from 10 to 13 hours per day. Consequently, we feel that a reasonably conservative assumption is 12 hours of Type F at 1 m/sec, 6 hours of E at 2 m/sec and 6 hours of D at 3 m/sec averaged over a 22½ degree sector.

For the 30-day release a maximum monthly sector wind direction frequency of 20 percent seems appropriate from the seasonal frequencies presented in Fig. 2A5. We also assumed the diffusion conditions to be equally distributed among Types F, D and C at 2, 3 and 3 m/sec respectively and averaged over a 22½ degree sector.

Based on a 15 percent prevailing wind direction frequency at Saginaw on an annual basis (Fig. 2A5) we assumed for the annual concentration a Type D condition with a 4 m/sec wind speed averaged over a 22½ degree sector.

In summary, a comparison of the ESSA computed relative concentrations at 500 m from a ground source to those calculated by the applicant are as follows:

2

Release	ESSA	Applicant
2-hr	7×10^{-4} sec m^{-3}	2×10^{-4} sec m^{-3}
24-hr	3×10^{-4} "	3×10^{-5} "
30 day	3×10^{-5} "	1×10^{-5} "
Annual	8×10^{-6} "	6×10^{-6} "

APPENDIX C

Comments on

Midland Plant Units 1 and 2
Consumers Power Company
Preliminary Safety Analysis Report
Amendment No. 6 dated December 29, 1969

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
February 4, 1970

We do not necessarily agree with the statement in Amendment 6 that the use of the Dow Chemical Building 47 wind data would be "conservatively representative of the site meteorology". As described in the PSAR, the wind system is on top of a 30-ft mast on the western edge of the flat roof of a 3-story building. Total height above the ground is 60 feet. The area from the northwest through northeast to the southeast is an entirely built-up urban area either of the city of Midland or the Dow Building complex itself. Also to the west and south, at least for a distance of several thousand feet to the Tittabawassee River, the area is within the Dow Building complex.

The site is essentially undeveloped marshland for 1500 feet in all directions and for many miles to the southwest, south and east the terrain is rural. The critical exclusion distance of 1500 feet is towards the north since the site boundary to the south is at least 1 mile. Thus, a ground release in the critical direction (to the north) would be carried by air having a rural trajectory. In contrast, the Dow data is taken 60 feet above the ground and presumably is affected by the turbulence generated by the rough and heated urban environment.

In addition to the reservations stated by the applicant with regard to the use of Dow data (see p. 2A-32, PSAR) and those reservations as stated in our comments of Feb. 3, 1969, we feel that a surface (10 meter height) measurement of wind over the marshy terrain of the site would be considerably more appropriate for site evaluation than the Dow Building 47 data.

APPENDIX D

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

EIS DOIGS

MAY 7 1970

Mr. Harold Price
Director of Regulation
U.S. Atomic Energy Commission
7920 Norfolk Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to a request by Mr. Roger S. Boyd, is a review of the geologic and hydrologic aspects of the Midland Plant Unit Nos. 1 and 2 - AEC Docket Nos. 50-329 and 50-330 - proposed by the Consumers Power Company.

This review was prepared by H. H. Waldron and P. J. Carpenter and has been discussed with members of your staff. We have no objections to your making this review a part of the public record.

Sincerely yours,

W. A. Radlinski

Acting Director

Enclosure

cc: Walter G. Belter, AEC

Midland Plant Units Nos. 1 & 2
Consumers Power Company

AEC Docket Nos. 50-329 and 50-330

The planned location for the Midland Plant Units Nos. 1 & 2 is on the south shore of the Tittabawassee River at the southern city limits of Midland, Midland County, Michigan. The plant is bounded on the west by Bullock Creek-drainage area, approximately 40 square miles. The plant will use two pressurized water nuclear reactors each rated at an ultimate output of 2,552 megawatts thermal, and a combined output of approximately 1,300 megawatts electrical. An artificial cooling pond will be used as a storage reservoir and as a heat sink for the condenser cooling water.

The following comments concerning the geology and hydrology of the site are based on an independent analysis of the data presented by the applicant in the "Preliminary Safety Analysis Report" and "Amendments" as well as an independent check of other available data and literature. The site was visited on August 14, 1969, and February 11, 1970. The analyses as presented by the applicant appear to adequately appraise those geologic and hydrologic conditions pertinent to the safety evaluation of the site.

Geology

The site is located in the Saginaw Lowland portion of the Great Lakes section of the Central Lowland physiographic province. At the site a crystalline basement complex is overlain by more than 12,000 feet of nearly flat-lying Paleozoic sedimentary rocks, and by about 360 feet of glacial sediments, chiefly fine-grained glacial lake deposits. Bedrock at the site consists of shale and interbedded sandstone and siltstone of the Saginaw Formation of Pennsylvanian age. The applicant proposes to excavate the upper layer of loose sand, which ranges in thickness from a few inches to as much as 35 feet, and to found the containment structure on the underlying very stiff to hard, preconsolidated, lacustrine clay unit, which ranges in thickness from 130 to 190 feet. All other major plant structures will be founded either on this hard clay unit or on compacted fill, or partly on both.

Tectonically the site is situated near the center of the Michigan Basin, a major regional structural basin that underlies the southern peninsula of Michigan and parts of adjoining states. Although there are no active faults or other recent geological structures known in the area that could be expected to localize seismicity in the immediate vicinity of the site, structural details in the underlying Paleozoic sedimentary rocks or in the crystalline basement complex are only very poorly known. Several structural features of a lesser magnitude have been mapped or have been postulated to exist within the Michigan Basin. Most of these features are ancient, northwest-trending anticlinal, synclinal, or monoclinal structures that have been delineated as a result of extensive oil and gas investigations. The site area appears to be located on one of these minor features locally known as the "Midland Trough"; the axial trace of the closest anticlinal structure approaches to within about 10 to 15 miles southwest of the site. Although normal faulting is reported to be associated with some of these structures, especially those in the southern part of the basin, none has been reported in the vicinity of the plant site. Most of the deformation apparently took place in early Paleozoic time. Deformation is greatly diminished to absent in the younger Paleozoic rocks, and none of these secondary features is known to extend to the surface or to have disrupted any of the glacial deposits in Michigan.

Natural brines and salt have been and still are being removed from a brine aquifer and from beds of salt that occur in the Detroit River Group (Devonian) at depths ranging from about 4,100 to 5,100 feet in the vicinity of the proposed nuclear power plant. The plant site overlies the projected eastern extremity of this brine and salt producing area. Although surface subsidence due to the extraction of natural brines appears to be precluded by the methods used in the extraction process, detailed studies and analyses by the applicant indicate that some very minor, broad, trough-type surface subsidence may occur in the site area due to solution mining of the salt beds; the effects at the actual plant site, however, will be very small, and surface rupture due to subsidence will not occur. In order to still further assure the safety of the plant, however, it is recommended that a precise monitoring system be installed for the purpose of detecting the occurrence of and determining the amounts of any possible future surface displacements that might occur due to subsidence in the plant area.

Hydrology

The plant grade will be established at elevation 634 feet above mean sea level. The stage for the Tittabawassee River probable maximum flood, as computed in 1956 for Dow Chemical Company, is given as 632 feet above mean sea level. The discharge of the computed probable maximum flood-270,000 cubic feet per second-included a flow of 20,000 cubic feet per second resulting from the breaching of four upstream low-head dams. This discharge is approximately 7.8 times greater than the maximum flood of record (34,800 cubic feet per second; March 28, 1916; stage, 610 feet above mean sea level) and is approximately 2.2 times greater than the maximum discharges observed for nearby, like-sized, drainage basins which appear to exhibit a similar extreme-flood potential. The applicant has independently reevaluated the probable maximum flood, dam breaching and the resultant stage at the plant site. The results of these reevaluations, as given by the applicant in discussions, are near but slightly below those originally presented. In addition, the applicant has evaluated the probable maximum flood and corresponding stage for Bullock Creek occurring simultaneously with a 100-year flood flow on the Tittabawassee River. The results of the reevaluations for Tittabawassee River and the Bullock Creek study were not made available in final form for our technical evaluation prior to this review. However, from discussions with the applicant it can be stated that the computational procedures used appear to be appropriate and the applicable hydrologic parameters appear to have been evaluated and applied correctly; if so, the final results should be reasonable. In any event, the applicant has stated that he "agrees to provide whatever additional flood protection, if any, is required in accordance with the revised probable maximum flood computations." It should be noted that the estimate of the probable maximum flood is the result of a theoretical calculation dependent on available meteorologic and hydrologic data. As more such data becomes available this estimate could be revised upward.

An emergency cooling pond will be constructed in an exacavation in the bottom of the operating cooling pond. The applicant states that if the operating cooling pond dikes were to fail the emergency pond could provide the amount of cooling water needed for 30 days of plant shutdown without make-up water from the Tittabawassee River. It is our understanding that after 30 days the required amount of make-up water would be less than 2 cubic feet per second. The minimum instantaneous flow observed on the Tittabawassee River (period of record, 1936 to 1966) was 39 cubic feet per second and the minimum daily flow was 111 cubic feet per second. Assuming that the integrity of the emergency cooling pond can be maintained in case of failure of the dikes of the operating cooling pond, an adequate supply of make-up water for safe shutdown of the plant appears to be assured.

The applicant has stated that operationally produced radioactive liquids will be released at maximum permissible concentrations specified in 10 CFR 20 to the Tittabawassee River and only under extreme or emergency conditions will radioactive liquids be discharged to the cooling pond. Further, such discharges to the pond will be at a level to insure that 10 CFR 20 concentration limits in the pond are not exceeded and prior to such discharges, detailed analyses of the potential radionuclide concentration and possible aquifer contamination will be made available for review and acceptance by the Division of Regulation, Atomic Energy Commission.

The applicant has stated that the radioactive waste system is contained within Class 1 structures, and accidental liquid releases due to component or piping failures would be contained within these structures. No estimates of the amount and composition of potential accidental radioactive liquid discharges have been made. The Tittabawassee River immediately downstream from Midland apparently is not used for domestic or municipal water supplies but is used mainly as an industrial water supply. The nearest downstream municipal water supply appears to be located in Saginaw Bay, some 40 to 50 miles downstream.

It should be mentioned that the Water Resources Commission, State of Michigan (1960) has stated that the water requirements in the Midland area for cooling, processing, and waste assimilation have already exceeded the supply. Consequently, all liquid waste discharges from the plant should be restricted to as low a level as is practically possible.

Assuming that future ground-water developments do not alter significantly the hydraulic gradients or head relationships in the aquifers under the site, ground-water supplies of the area should not be affected by accidental spills of radioactive liquids for the following reasons: (1) the hydraulic gradients in the shallow water-table aquifer, as determined by borehole observations, are toward the Tittabawassee River, (2) all domestic wells dug or drilled into the water-table aquifer in the area apparently are located upgradient of the site, (3) a relatively impermeable clay layer, some 130 to 190 feet thick, separates the water-table aquifer and the underlying artesian aquifer which furnishes potable water supplies, (4) data from a pump test show that the piezometric surface of the artesian aquifer is above the water table, and (5) all wells in the cooling pond area will be effectively filled and sealed by the applicant to prevent fluids from entering the aquifers directly.

<u>Reference</u>:

Water Resources Commission, State of Michigan, 1960; Water Resources Conditions and Uses in the Tittabawassee River Basin.

APPENDIX E

U.S. DEPARTMENT OF COMMERCE
Environmental Science Services Administration
COAST AND GEODETIC SURVEY
Rockville, Md. 20852

Reply to
Attn of: C23

MAY 7 1970

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

Dear Mr. Price:

In accordance with your request, we are forwarding 10 copies of our report on the seismicity of the Midland, Michigan, area. The Coast and Geodetic Survey has reviewed and evaluated the information on the seismic activity of the area as presented by the Consumers Power Company in the "Preliminary Safety Analysis Report," for use in the evaluation of the site of the proposed Midland Nuclear Power Plant, Units 1 and 2; and we hereby submit our conclusions concerning the seismicity factors.

If we may be of further assistance to you, please contact us.

Sincerely,

for Don A. Jones
Rear Admiral, USESSA
Director, C&GS

10 Enclosures

REPORT ON THE SITE SEISMICITY

FOR THE MIDLAND NUCLEAR

POWER PLANT UNITS 1 & 2

At the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismology Division of the Coast and Geodetic Survey has evaluated the seismicity of the area around the proposed Midland Nuclear Power Plant near Midland, Michigan. The Survey has also reviewed a similar evaluation presented by the Consumers Power Company in their "Preliminary Safety Analysis Report."

Historically, very few earthquakes have occurred in the vicinity of this plant site. However, two intensity VI (MM) earthquakes and several smaller events have occurred within 150 miles of this site. Consideration is also given to the major, although distant, earthquakes that may have affected this site. These events include the very large earthquakes in the St. Lawrence region and the 1811-1812 earthquakes at New Madrid, Missouri.

The first of the historical intensity VI (MM) earthquakes occurred on February 4, 1883 and caused damage to glass in Kalamazoo, Michigan. The second intensity VI event occurred on August 9, 1947 and damaged chimneys and plaster at Athens, Coldwater, Colon, Matteson Lake, Sherwood, and Union City, Michigan. Both of these events have epicenters over 100 miles from this plant site.

Since the major earthquake regions, such as New Madrid, Missouri and the St. Lawrence area, are over 400 miles away, they are not considered to have a significant affect on the determination of the acceleration factor for this site.

As reported by the U. S. Geological Survey to the Atomic Energy Commission this site is located in the Saginaw portion of the Great Lakes section of the central lowland physiographic province and the tectonic region of the Michigan Basin. This report also states that there are no active faults or other recent geological structures known that could be expected to localize seismicity in the immediate vicinity of the site. But the report also states that "structural details in the underlying Paleozoic sedimentary rocks or in the crystalline basement complex are only very poorly known." Therefore, it must be assumed that earthquakes with intensities comparable to the earthquakes that have occurred in the Michigan Basin might also occur in the vicinity of the plant site.

In further consideration of the earthquake intensities, it is noted that this plant is to be located on a clay formation which is the upper part of the glacial sediments of approximately 360 feet thickness which in turn overlie approximately 12,000 feet of Paleozoic sedimentary rocks.

The salt extraction activity in the area is not considered to have a significant effect on the calculation of the seismic factors since the extraction zones are at great depth and since a brine recharging program is practiced.

As a result of this review of the seismological and geological characteristics of the area around the plant site, the Coast and Geodetic Survey recommends that an acceleration of 0.06 g, resulting from an intensity V (MM) earthquake, would be adequate for representing seismic disturbances likely to occur within the lifetime of the facility. The Survey also recommends that an acceleration of 0.12 g, resulting from an intensity VI (MM) earthquake, would be adequate for representing the ground motion from the maximum earthquake likely to affect the site. It is believed that these values would provide an adequate basis for designing protection against the loss of function of components important to safety.

U. S. Coast and Geodetic Survey
Rockville, Maryland 20852

May 5, 1970

APPENDIX F

United States Department of the Interior

OFFICE OF THE SECRETARY
WASHINGTON, D.C. 20240

MAR 27 1969

Dear Mr. Price:

This will transmit the comments of the Fish and Wildlife Service on the application by Consumers Power Company for a construction permit and facility license for the proposed Midland Generating Plant, Units 1 and 2, Tittabawassee River, Midland County, Michigan, AEC Dockets Nos. 50-329 and 50-330. These comments are provided in response to Mr. Boyd's letter of November 7, 1968.

The project would be located adjacent to the southern boundary of Midland, Michigan, on the south bank of the Tittabawassee River and would use two pressurized water reactors, each designed for an initial output of 2,452 megawatts thermal and a gross electrical output of 650 megawatts.

The condensers would be cooled by water recirculated from a 14,000 acre-foot storage pond constructed on the flood plain at the plant site. The applicant proposes to fill the pond during the flood season since pumping at this time requires a smaller pumping head and there is less chance of reducing the residual river flow to undesirable low levels. After initial filling of the pond, makeup water would be required at a rate of approximately 70 c.f.s. to maintain a full storage pool. The storage pond would have the capacity to supply the plant for about 100 days without the addition of makeup water. Therefore, no pumping would be undertaken during periods of insufficient river flow.

The water quality in the project area is significantly lowered by the addition of industrial pollutants from nearby plants. However, the river supports a moderate sport fishery for largemouth bass, yellow perch, bluegill, carp, catfishes, and suckers.

The application indicates that the release of radioactive wastes would not exceed limits prescribed in title 10, part 20, of the Code of Federal Regulations. If the concentration in the receiving water were the only consideration, maximum permissible limits would be adequate criteria for determining the safe rate of discharge for fish and wildlife. However, radioisotopes of many elements are concentrated and stored by organisms that require these elements for their normal metabolic activities. Some organisms concentrate and store radioisotopes of elements not normally required but which are chemically similar to elements essential for metabolism. In both cases, the radionuclides are transferred from one organism to another through various levels of the food chain just as are the nonradioactive elements. These transfers may result in further concentration of radionuclides and a wide dispersion from the project area, particularly by migratory fish, mammals, and birds.

It is imperative that every possible effort be made to protect the fish and wildlife resources of the area from radioactive contamination. An environmental radiological monitoring program is needed to determine if the radionuclides released to the environment are affecting fish and wildlife resources adversely. This program should be planned in cooperation with the Fish and Wildlife Service and the Michigan Department of Natural Resources.

In order to provide for the conservation, development, and protection of the fish and wildlife resources, it is recommended that Consumers Power Company be required to:

1. Cooperate with the Fish and Wildlife Service, the Michigan Department of Natural Resources, and other interested State and Federal agencies in developing plans for radiological surveys.

2. Conduct pre-operational radiological surveys including but not limited to the following:

   a. Gamma radioactivity analysis of water and sediment samples collected within 500 feet of the reactor effluent outfall.

   b. Beta and gamma radioactivity analysis of selected fish and wildlife species and organisms important in their food chain collected as near the reactor effluent outfall as possible.

3. Prepare a report of the pre-operational radiological surveys and provide 5 copies to the Secretary of the Interior for evaluation prior to project operation.

4. Conduct post-operational radiological surveys similar to those specified in recommendation 2 above, analyze the data, prepare reports every 6 months during reactor operation until it has been conclusively demonstrated that no significant adverse conditions exist, and submit 5 copies of these reports to the Secretary of the Interior for distribution to appropriate State and Federal agencies for evaluation.

5. Make such reasonable modifications of project structures and operations as may be ordered by the Atomic Energy Commission upon its own motion or upon the recommendation of the Secretary of the Interior or the Michigan Department of Natural Resources, after notice and opportunity for hearing and upon findings that such modifications are necessary and desirable.

We understand that the regulatory authority of the Atomic Energy Commission is confined to considerations of common defense, security, radiological health, and safety. However, we recommend and urge that before the permit is issued, the dangers of other potential hazards to fish and wildlife resources which may result from plant construction and operation be called to the attention of the applicant. Sufficient numbers of fish may be drawn into the intake to reduce the fishable population in the Tittabawassee River below desirable levels. The release of plant wastes, coupled with the anticipated reduction of the flow of the river, may create further hazards to aquatic life. The applicant should meet with representatives of the Fish and Wildlife Service, the Federal Water Pollution Control Administration, and the Michigan Department of Natural Resources to discuss these and any other hazards and should jointly design means to monitor project effects and to mitigate conditions found adverse to fish and wildlife resources.

In view of the Administration's policy to maintain, protect, and improve the quality of our environment, we request that the Commission urge Consumers Power Company to:

1. Cooperate with the Fish and Wildlife Service, the Federal Water Pollution Control Administration, and the Michigan Department of Natural Resources in designing measures to monitor the effects of the project on the natural resources of the area.

2. Take such steps as may be determined necessary by the above named agencies to mitigate any adverse effects of the project.

The opportunity for presenting our views is appreciated.

Sincerely yours,

Charles C. Crowthers III

Deputy Assistant Secretary of the Interior

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

50-329
50-330

APPENDIX G

REVIEW OF THE SEISMIC DESIGN CRITERIA

FOR THE

MIDLAND PLANT

(Docket No. 50-329 and 50-330)

May 5, 1970

JOHN A. BLUME & ASSOCIATES, ENGINEERS
San Francisco, California

REVIEW OF THE SEISMIC DESIGN CRITERIA

FOR THE MIDLAND PLANT

(Docket No. 50-329 and 50-330)

This report summarizes our review of the engineering factors pertinent to the seismic and structural adequacy of the Midland Plant. The plant is located along the south shore of the Tittabawassee River adjacent to the Dow Chemical Company's main complex in Midland, Michigan. The design and construction of the plant will be performed by Bechtel Corporation under direction of the applicant, Consumers Power Company. The nuclear steam supply system will be supplied by the Babcock & Wilcox Company. The plant will be composed of two units having a combined capability of 1,300 MWe and 4,050,000 lb/hr of process steam. The process steam will be supplied to the Dow Chemical Company and the electricity to the applicant. Application for a construction permit has been made to the U. S. Atomic Energy Commission (AEC Docket Nos. 50-329 and 50-330) by Consumers Power Company. A Safety Analysis Report has been submitted in support of the application to show that the plant will be designed and constructed in a manner which will provide for safe and reliable operation. Our review is based upon the information presented in the Safety Analysis Report and is directed specifically towards an evaluation of the seismic and structural design criteria for Class I structures, systems, and components. The list of reference documents upon which this review has been based is given at the end of this report.

DESCRIPTION OF THE FACILITY

The Midland Plant site is located on a level plain formed by glacial lake deposits. Elevations vary from about 600 ft to 625 ft above mean sea level. Drainage is to the northeast into the Tittabawasse River. The river flows to the southeast and coincides with the northeast boundary of the site. The uppermost soil in the area is quartz sand which is locally clayey and varies from 0 to 40 ft in thickness. Below this sand is a layer of blue-gray clay which in turn is underlain by sands and gravels to a total depth of about 350 ft. These unconsolidated Pleistocene glacial lake deposits rest unconformably upon well consolidated sediments of Pennsylvania age. The reactor and auxiliary buildings will be supported on mat foundations on the clay layers underlying the uppermost sand. This material varies from stiff to hard and should provide adequate support. Other major structures will be founded partly or entirely upon compacted fill.

The containment structure will be a prestressed concrete cylinder and dome which will be supported on a reinforced concrete foundation slab. The interior of the structure will be lined with a 1/4-inch thick welded steel plate to ensure leak tightness. The inside diameter of the containment structure will be 116 ft and the inside height including the dome will be 193 ft. The vertical wall thickness will be 3-1/2 ft and the dome thickness will be 3 ft. The foundation slab thickness will be 9 ft. The dome and walls of the containment structure will be post-tensioned. This post-tensioning system will consist of three groups of dome tendons oriented at 120° to each other and anchored at the vertical face of the dome ring girder; the walls are to be post-tensioned by vertical tendons anchored at the top surface of ring girder and at the bottom of the base slab. In addition, three groups of hoop tendons enclosing 240° of arc will be anchored at three vertical buttresses.

STRUCTURAL DESIGN CRITERIA AND LOADS

All structures, equipment, systems, and piping are classified according to function or consequence of failure as either Class 1 or 2 as defined in Appendix 5A of the Safety Analysis Report. Class 1 structures, systems, and equipment are those whose failure could cause uncontrolled release of radioactivity or are those essential for immediate and long-term operation following a loss-of-coolant accident. They are designed to withstand the appropriate seismic loads simultaneously with other applicable loads without loss of function. Class 2 structures, systems, and equipment are those whose failure would not result in a release of radioactivity and would not prevent reactor shutdown but may interrupt power generation.

The design loads for the Midland Plant are divided into two basic categories. The first category includes normal operation (dead, live, and prestress loads) and the second category includes accident, seismic and tornado conditions. Structure design loads will be increased by load factors based on the probability and conservatism of the predicted design loads. Yield capacity reduction factors will be applied to the stresses allowed by the applicable building codes.

ADEQUACY OF THE SEISMIC DESIGN CRITERIA

We have reviewed the Preliminary Safety Analysis Report and Amendments No. 1 through 10 and have discussed the various aspects of the seismic design of the plant with the applicant and members of the staff of the Division of Reactor Licensing at meetings on January 29, 1970, and March 19, 1970. We have the following comments regarding the adequacy of the seismic design criteria:

1. The data submitted by the applicant has included detailed discussions and analyses of allowable bearing pressures, settlements in founding materials, and the possibility of liquefaction.

2. According to data submitted by the applicant, there is no known faulting near the site. The nearest faulting is about 55 miles south of the site consisting of a questionable fault zone which probably trends northwesterly. Other faults are known which are situated 325 miles northwest and 240 miles northwest of the site. The low dipping basement rocks contain gentle folds but are otherwise relatively undisturbed. The condition of the Paleozoic basement rocks indicates that the region has not been subjected to significant tectonic activity since at least the Paleozoic Era.

3. Midland, Michigan is in a seismically quiet area. Five earthquakes are known to have been centered within 150 miles of the site, and none of these were strongly at Midland. There is no known geologic control of earthquake occurrence or distribution in the region. The greatest historic shock felt at Midland is estimated to have had a MM intensity V or an equivalent acceleration of about 0.03g. A value of 0.06g maximum ground acceleration is postulated for the "Design" Earthquake and 0.12g is postulated for the "Maximum" Earthquake. We concur with the selection of these ground accelerations. The site response spectra for the Design and Maximum Earthquakes and the application of these site spectra, including provisions for safety margins, as proposed by the applicant in Amendment 10, pages 9.00-3 and 900-4 are satisfactory and if properly implemented will result in a conservative design.

4. The applicant has stated that he will use the response spectrum method of dynamic analysis for Class I structures, piping, and equipment. The structures will be analysed for response in both the horizontal and vertical directions, and a range of foundation material moduli will be used in the analyses to account for variations in these moduli. Time-history analyses of Class I structures will be performed to develop response spectra in vertical and horizontal directions at the points of support of piping and equipment.

   The applicant has proposed to analyse some piping systems for a static load equal to the peak of the response spectrum curve at points of support of the system in lieu of performing a dynamic analysis. This method will be used only when twice the resulting seismic stresses in combination with other applicable stresses are below the code allowable stress. The applicant has presented representative comparisons of the results of analyses utilizing the proposed static loading approach and the results of dynamic analyses of the same systems which demonstrate the conservatism of the proposed approach.

We concur in general with the proposed approach to the seismic design of Class I structures, piping, and equipment. The analytical techniques proposed by the applicant are satisfactory and if properly implemented will result in a conservative design.

CONCLUSIONS

On the basis of the information presented by the applicant in the Preliminary Safety Analysis Report and Amendments, it is our opinion that the seismic design criteria and approach to seismic design as outlined in the PSAR and Amendments 1 through 10, if properly implemented by the applicant, will result in a design that is adequate to resist the earthquake conditions postulated for the site.

JOHN A. BLUME & ASSOCIATES, ENGINEERS

Roland L. Sharpe

Roland L. Sharpe

Garrison Kost

Garrison Kost

REFERENCES

MIDLAND PLANT

CONSUMERS POWER COMPANY

Preliminary Safety Analysis Report, Volumes I, II, and III
Amendments 1 through 10
"Midland Nuclear Site Considerations"

APPENDIX H

AEC REGULATORY STAFF'S EVALUATION OF THE FINANCIAL QUALIFICATIONS OF CONSUMERS POWER COMPANY

DOCKET NOS. 50-329 AND 50-330

We have reviewed the financial information in the application, amendment no. 13 and in the 1969 and previous Annual Reports of the Consumers Power Company for a permit to construct two nuclear reactors with an initial thermal power level of 2,452 Mwt each to be known as the Midland Plant Units Nos. 1 and 2 and to be located in Midland Township, Midland County, Michigan. Based on this information, we have concluded that the Consumers Power Company (Consumers) is financially qualified to design and construct the proposed Midland Plant Units 1 and 2 (Midland). This conclusion is based upon the following facts and considerations:

1. The applicant estimates the costs of construction of Midland, including first core fuel cost for each unit, to be $394,827,000, made up as follows:

Total nuclear production plant costs	$346,640,000
Transmission, di_ ibution and general plant costs	3,145,000
Nuclear fuel inventory for first cores	45,042,000
Total	$394,827,000

The above estimate of construction costs (pages 5 and 6 of amendment no. 13 dated June 2, 1970) contains allowances for escalation, scope changes and contingencies. The details of these estimates as they pertain to the capital costs of the nuclear plant have been reviewed by the Division of Reactor Licensing and found to be reasonable. The Division of Reactor Development and Technology has reviewed the fuel requirements specified by the applicant as 207,486 pounds of UO_2 for the first core of each unit of the Midland plant, and finds them reasonable for reactors of this type and power level.

2. The Midland plant is a necessary part of applicant's continuing expansion of its facilities to provide for the steadily increasing demand for electric power by its customers. The applicant will use the plant for this purpose as well as to provide some of the process steam generated to a customer (Dow Chemical Co.) on an adjacent site. The entire cost of the project will be paid for by Consumers from funds available from normal and regular sources for construction of additions to all types of its utility properties. Such funds are obtained from funds internally generated, principally unappropriated earnings and provisions for depreciation, short-term loans and the sale of debt and equity securities when and as required.

3. Based on Consumers' record of earnings and provision for depreciation and other accruals over the past six years, on the reasonable assumption of the continuation of relatively the same level of earnings

over the next six years, and in view of Consumers' resources, the strength of its financial position, the very high regard held for its bond issues and its proven ability to borrow on a short-term basis, it is our opinion that the above-stated sources can be relied upon with reasonable assurance to supply the funds required over the next several years, as set forth in amendment no. 13 and Appendix A thereto, to design and construct the Midland nuclear plant.

4. Consumers is soundly financed and has significant resources at its command. As of December 31, 1969 cash and net receivables totaled $57.6 million. Operating revenues totaled about $550 million for the year. The long-term debt represented 55.6% of total capitalization and the company is not overcapitalized on a book value basis as evidenced by the ratio of net plant to capitalization of 1.13. The company's Dun and Bradstreet credit rating is AaA1 and its Moody's Investors Service mortgage bond rating is Aaa (blue chip).

Operating revenues have increased over the past six years from $376.4 million in 1964 to $549.8 million in 1969 or over 46%. The pertinent financial ratios for CY 1969 (and previous years) indicate a sound financial position and are in line with those of the electric utilities as a whole. A copy of the financial analysis of this company reflecting these ratios and other pertinent data is attached (Attachment "A").

Attachment "A"

CONSUMERS POWER COMPANY
FINANCIAL ANALYSIS
Docket Nos. 50-329 and 50-330

	(dollars in millions) Calendar Year Ended Dec. 31			
	1969	1968	1967	1964
Long-term debt	$ 810.6	$ 714.6	$ 612.0	$ 486.6
Utility plant (net)	1,643.0	1,496.3	1,344.1	1,081.6
Ratio - debt to fixed plant	.49	.48	.46	.45
Utility plant (net)	1,643.0	1,496.3	1,344.1	1,081.6
Capitalization	1,458.4	1,331.8	1,208.8	1,012.8
Ratio to net plant to capitalization	1.13	1.12	1.11	1.1
Stockholders' equity	647.9	617.2	596.8	526.2
Total assets	1,808.5	1,640.7	1,477.4	1,201.6
Proprietary ratio	.36	.38	.40	.44
Net income	67.0	62.6	68.5	54.0
Stockholders' equity	647.9	617.2	596.8	526.2
Rate of return on stockholders' investment	10.3%	10.1%	11.5%	10.3%
Net income before interest	97.7	88.1	91.3	71.8
Liabilities and capital	1,808.5	1,640.7	1,477.4	1,201.6
Rate of return on total investment	5.4%	5.4%	6.2%	6.0%
Net income before interest	97.7	88.1	91.3	71.8
Interest on long-term debt	36.0	29.0	23.6	17.3
No. of times fixed charges earned	2.7	3.0	3.9	4.2
Operating expenses (including taxes)	454.9	419.4	387.3	306.3
Operating revenues	549.8	505.1	477.2	376.4
Operating ratio	.83	.83	.81	.81
Utility plant (gross)	2,111.0	1,924.1	1,742.0	1,383.9
Operating revenues	549.8	505.1	477.2	376.4
Ratio of plant investment to revenues	3.83	3.81	3.65	3.68
Retained earnings	156.5	125.5	109.6	134.0
Earnings per share of Common	$2.79	$2.59	$2.87	$2.46

	1969		1968	
Capitalization as of Dec. 31	Amount	% of Total	Amount	% of Total
Long-term debt	$ 810.6	55.6%	$ 714.6	53.7%
Preferred stock	79.1	5.4	79.6	6.0
Common stock	568.7	39.0	537.6	40.3
Total	$1,458.4	100.0%	$1,331.8	100.0%

Moody's Bond Ratings:	
First Mortgage Bonds	Aaa
Debentures	Aa
Dun and Bradstreet Credit Rating	AaA1