50-329 ACRS REPORT, MIDLAND PLANT UNITS NOS. 1+2

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Docket Nos. 50-329 &
50-330

March 6, 1970

Report to ACRS

MIDLAND PLANT UNIT NOS. 1 & 2

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U. S. Atomic Energy Commission
Division of Reactor Licensing

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ABSTRACT

The Consumers Power Company has submitted an application for a construction permit for the two-unit Midland Plant. The nuclear steam supply systems are very similar to other B&W plants which we have reviewed and found acceptable, such as Three-Mile Island 2, Arkansas Nuclear One, and Rancho Seco. The initial power level of the facility is 2452 Mwt, with an anticipated ultimate power capability of 2552 Mwt.

The plant site, which is on the right bank of the Tittabawassee River, is located adjacent to the Dow Chemical Company complex. Considering industrial population, the cumulative population within 5 miles of the facility is higher than for any previously approved power reactor. Solution salt mining is being conducted beneath the site. We are continuing to evaluate the degree of subsidence which may result.

We conclude that the proposed plant can be constructed and operated without undue risk to the health and safety of the public provided (1) the potential for significant surface subsidence is found to be acceptably small, (2) analyses of the stability of plant fill slopes demonstrate an acceptable factor of safety, (3) adequate flooding protection is provided, (4) an adequate onsite meteorological program is developed, (5) the control room ventilation system is shown to be adequate to prevent high levels of toxicity following an accidental release at Dow Chemical Company, (6) confirmatory preoperational vibration tests are performed, (7) the cooling pond dike is designed to withstand flooding or the service water system bar screens are designed to cope with flood debris, (8) diversity is provided in ECCS initiation signals, (9) the pressurizer high-level-alarm system is designed to reactor protection system standards, (10) the control room design is shown to limit post-accident doses to acceptable levels, using staff assumptions, (11) an adequate system is developed to monitor activity in the export process steamlines and (12) adequate plans are developed to investigate means of coping with hydrogen evolution following a loss-of-coolant accident.

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1.0 INTRODUCTION AND SUMMARY

1.1 General

On January 13, 1969, the Consumers Power Company filed an application for a construction permit for two nuclear power units to be located at the Midland site on the southern boundary of Midland, Michigan, on the right bank of the Tittabawassee River. The nuclear steam supply systems will be supplied by The Babcock & Wilcox Company and will each initially operate at power levels up to 2452 megawatts thermal (Mwt). The ultimate capability of each unit is 2552 Mwt. 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 exported to the adjacent Dow Chemical Company complex where it will be used in the production of industrial chemicals and pharmaceuticals. Under normal operating conditions Unit No. 1 will supply 492 Mwe (net), and 3.65×10^6 pounds per hour of 197 psia process steam and 4×10^5 pounds per hour of 675 psia process steam to Dow Chemical. Unit No. 2 will normally supply 818 Mwe (net). Under circumstances where Unit No. 1 is not in service, Unit No. 2 will supply 380 net Mwe and process steam as described above for Unit No. 1.

The location and population distribution of the site were considered by the ACRS at its meeting on February 6, 1969. At the conclusion of the ACRS meeting with the applicant, the Committee indicated that the proposed site was unacceptable for use with reactor plants designed and analyzed as described in the PSAR. However, the Committee stated it believed the site may be acceptable for use with reactor plants of the proposed power rating if:

1. The facility is equipped with adequate engineered safety features and protective systems;
2. The facility is analyzed sufficiently conservatively, particularly in respect to determination of exclusion area and low population zone, assurance of low potential doses at short distances from the reactor in the unlikely event of a serious accident, evaluation of the number and location of people who could be safely and quickly evacuated in such an event, and use of assumptions; for example, those related to meteorology, in dose calculations;
3. The facility is designed, constructed, and utilized sufficiently conservatively; and
4. The facility is provided with thoroughly structured effective emergency plans including evacuation plans.

1.2 Plant Modifications Since Site Review

As a result of the Committee's comments and our discussions with Consumers, the applicant has made the following design changes in the Midland Plant:

1. Engineered Safety Features

   a. Chemical additives, sodium thiosulfate and sodium hydroxide, were added to the containment spray system.

   b. The design was modified to incorporate a penetration pressurization system and an isolation valve seal water system.

   c. Charcoal filters have been added to the auxiliary building ventilation system and the system has been modified such that fuel storage pool exhaust ventilation will be passed through charcoal filters when refueling.

   d. Pressurized weld channels or their equivalent will be added over the seam welds in the containment liner-plate.

   e. A sealed compartment has been provided to accommodate a failure in the emergency core cooling system suction line between the containment sump and the first isolation valve located outside containment.

   f. Provisions for a post-loss-of-coolant accident reactor vessel cavity flooding system will be added to the design.

   g. The applicant agreed to study flame and catalytic hydrogen recombiners, and a purge system was added to the containment ventilation system.

   h. The design leakage rate for the containment was reduced from 0.2% per day to 0.1% per day.

2. Site Related Items

   a. The exclusion area radius was reduced from approximately 2400 meters to 500 meters.

   b. The low population zone distance was reduced from 3 miles to 1600 meters.

   c. The applicant has agreed to establish an onsite meteorological program.

   d. The design basis accelerations for the operating basis earthquake and the design basis earthquake were increased to 0.06g and 0.12g, respectively.

2. Mechanical and Structural Design

   a. The primary system piping code was changed from USASI B31.1 to USASI B31.7, (ANSI B31.7).

   b. The Cadweld splice testing program was modified to include production splices.

   c. The control room shielding was redesigned to limit the whole body dose to personnel in the 30 days following a loss-of-coolant accident to 5 rem and the thyroid dose to 30 rem. (See discussion of assumptions in Section 7.5.1, however).

   d. All portions of the decay heat removal systems will be designed to seismic Class I specifications.

   e. The gaseous waste storage tanks will be designed to seismic Class I specifications.

f. The fire protection system has been redesigned so that failure of seismic Class II portions of the system will not prevent the functioning of seismic Class I equipment.

4. Protection System and Electrical Power

a. The nuclear instrumentation and protection system was modified to provide greater separation between control and protection channels.

b. The ac emergency power system was modified to eliminate the potential for connecting the two diesels to a single bus.

c. An additional 138 kV line has been added to provide another source of offsite power to the site.

5. Miscellaneous

a. Pressure vessel irradiation surveillance specimens will be placed in Unit No. 2.

b. The applicant agreed to monitor the air ejector offgas continuously.

c. The applicant agreed to establish a technical specification limit requiring verification that the pressurizer level is below a maximum value prior to withdrawing control rods.

1.3 Ma[j]or Areas of Review

Because of the similarity of the nuclear steam supply systems (NSSS) of the Midland Plant units with those of other B&W designed NSSS, our review has been based to a large extent on comparisons with these plants. Features which were significantly different were identified and evaluated in detail. Emphasis was placed on (1) upgrading the engineered safety features to provide protection commensurate with that provided at other high population-density sites, and (2) unique features associated with the site. A chronology of the principal events during the review is given in Table 1.3.

TABLE 1.3

CHRONOLOGY

REGULATORY REVIEW OF THE CONSUMERS POWER COMPANY MIDLAND PLANT UNITS 1 AND 2

1.	October 30, 1968	Consumers Power Company informally submitted its Preliminary Safety Analysis Report (Formal Application for Licenses not received).
2.	November 27, 1968	Meeting with applicant to discuss scheduling of regulatory review of application, AEC interpretation of FDA requirements on releases of radioactivity into foods.
3.	December 17, 1968	Meeting with applicant to discuss the Midland site related problems, such as population distribution, evacuation procedures, etc.
4.	January 13, 1969	Consumers Power Company formally filed Application for Licenses for the Midland Plant, Units 1 and 2.
5.	January 22, 1969	AEC-DRL staff and ACRS Subcommittee visited site of Midland Plant.
6.	January 23, 1969	AEC-DRL Report to ACRS concerning the Midland Plant exclusion area, low population zone, population distribution, meteorology, and accident analysis.
7.	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."
8.	February 4, 1969	ACRS Subcommittee Meeting to discuss suitability of Midland Plant site.
9.	February 5, 1969	Meeting with applicant to discuss continuing meteorological studies.

10.	February 6, 1969	ACRS meeting with applicant to discuss population distribution problem within exclusion area.
11.	March 21, 1969	Meeting with applicant to discuss applicant's plans for Midland Plant in light of ACRS meeting held on February 6, 1969.
12.	March 28, 1969	AEC-DRL staff notifies applicant that site is unacceptable for use with reactor plant design proposed and that a plant with adequate safety features is not precluded.
13.	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.
14.	July 15, 1969	Meeting with applicant to discuss the design review schedule for the Midland Plant.
15.	July 24, 1969	Meeting with applicant to discuss containment structural design and site geology.
16.	August 13, 1969	Submittal of Amendment No. 3, supplement to the Dames and Moore Foundation Investigation Report submitted by Amendment No. 1 to the PSAR.
17.	September 11, 1969	Submittal of letter by applicant expressing concern regarding slippage in schedule.
18.	September 18, 1969	AEC-DRL notifies applicant that AEC review is being conducted with full knowledge of scheduling needs.
19.	September 26, 1969	Request to applicant for additional information on site, reactor design, reactor coolant system design, structural design, engineered safety features and other miscellaneous items.

20.	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.
21.	October 30, 1969	Meeting with applicant to discuss Quality Assurance Program.
22.	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 and October 28, 1969.
23.	December 5, 1969	Meeting with applicant to discuss Amendment No. 5.
24.	December 16, 1969	Meeting with applicant to discuss COPATTA Code.
25.	December 29, 1969	Submittal of Amendment No. 6, revises and supplements information in PSAR and portions of information submitted by Amendment No. 5.
26.	January 8, 1970	Request to applicant for additional information on reactor site, design, coolant system design and miscellaneous other topics.
27.	January 20, 1970	Meeting with applicant to discuss Quality Assurance, meteorology, emergency power and tornado design.
28.	January 30, 1970	Submittal of Amendment No. 7, revised pages, amending the responses given in Amendments 5 and 6 and response to the AEC regulatory staff's request for additional information dated January 8, 1970, except item 2.13 on site data.
29.	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.

1.4 Summary

At the present time, there are three items for which we have not yet completed our review, five items for which we are in disagreement with the applicant, and several other items for which we will require additional information from the applicant during detailed design and construction of the facility. These items are identified below and the section of this report in which they are discussed is indicated. Subject to satisfactory resolution of these matters, we conclude that the proposed facility can be constructed and operated at the proposed site without undue risk to the health and safety of the public.

1.4.1 Items for Which Review is not Complete

1. Effects of salt mining on the potential for subsidence (Section 2.3).
2. Slope stability analyses to demonstrate factor of safety for plant fill slopes (Section 2.4).
3. Establishment of probable maximum flood levels (Section 2.6).

1.4.2 Items at Issue

1. Protection of control room occupants from release of toxic chemicals at the Dow Chemical Plant (Section 2.8). The applicant maintains that the proposed design, including provisions for use of Scott Air-Paks, provides adequate protection. We plan to verify the adequacy of the control room ventilation filtration design.

2. Vibration Testing (Section 3.1.4). The applicant plans minimum vibration testing for the Midland Plant because of the similarity between the Midland units and earlier B&W plants which will be subject to vibration testing. We plan to require confirmatory vibration testing as part of the Midland preoperational test program.
3. Diversity of ECCS Initiation Signals (Section 3.8). The applicant maintains that the reliability of the reactor coolant system low pressure ECCS initiation signal makes the need for a diverse high containment pressure signal for ECCS initiation and reactor trip unnecessary. We plan to require diverse ECCS actuation signals, either one of which will provide a reactor trip to assure the effectiveness of the emergency core cooling system.
4. Pressurizer High-Level Alarm (Section 7.1.1). The applicant dces not intend to install a pressurizer high-level trip and has not addressed the design criteria for the level instrumentation. We plan to require that the pressurizer high-water-level alarm system be designed to reactor protection system standards.
5. Protection of Control Room Occupants Following a LOCA (Section 7.5.1). The applicant maintains that doses in the control room following a LOCA should be calculated assuming (1) release of gap activity only, (2) iodine removal using the spray removal constant calculated by the applicant, (3) 5% methyl iodide, and (4) the applicant's meteorological model. We plan to require the use of source terms, a meteorological model and iodine removal assumptions consistent with other current staff reviews.

1.4.3 Additional Information to be Provided During Construction

1. Adequacy of the process steam monitor to detect leakage of primary coolant activity through the steam generator into the export process steam (Section 6.0).
2. Plans to control the hydrogen concentration in the containment following a loss-of-coolant accident (Section 7.5.2).
3. Onsite meteorological data to support the applicant's meteorological assumptions (Section 2.2).
4. Information to determine the adequacy of the cooling pond dike to withstand flooding without breeching or overtopping or to show that cooling water intakes are protected from flood debris (Section 3.6).

2.0 SITE

2.1 Site Location and Population Distribution

The location of the site and the population distribution in the vicinity of the site are discussed in our report to the Committee dated January 23, 1969. As indicated therein, the population in the vicinity of the Midland site is significantly higher than that for any previously approved site. The cumulative population in the immediate vicinity of the site exceeds that of Zion and Indian Point to distances of 5.7 and 6.2 miles, respectively. The cumulative population as a function of distance at the Zion, Indian Point, and Burlington sites is compared with the Midland cumulative population, including Dow Chemical Company employees, in Figure 2.1. If the population data are adjusted downward to take into account the fact that both business and residential populations are included in the total (and thus some people are counted twice), the population at the Midland site exceeds that of Zion and Indian Point to distances of 5.3 and 5.7 miles, respectively. Thereafter, the Zion and Indian Point populations exceed that of Midland by a substantial margin, because of the general agricultural utilization of the land around the city of Midland.

The applicant has modified his earlier proposal that the entire Dow Company complex be within the exclusion area. He also reduced the calculated offsite accident doses by decreasing containment leakage and by

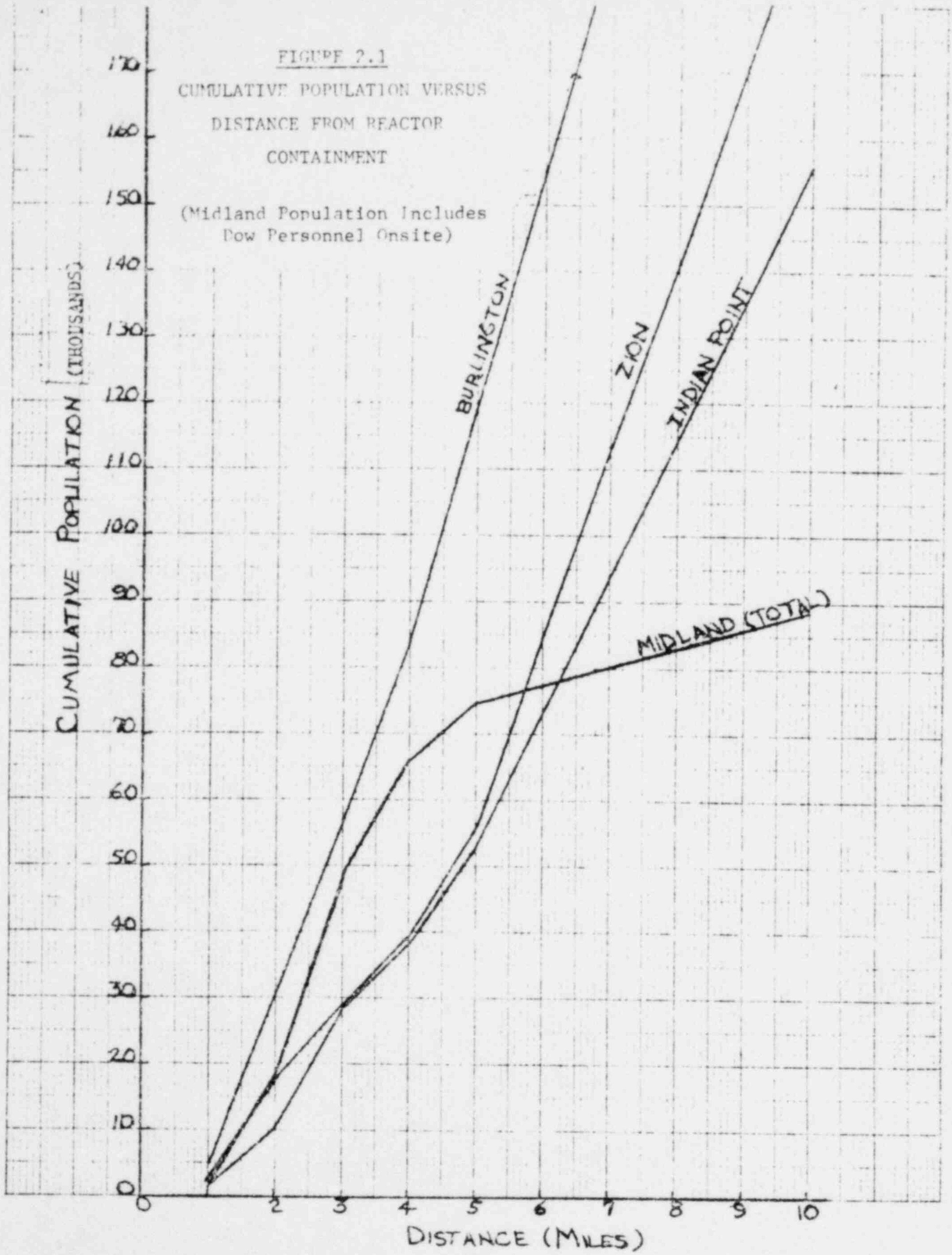
FIGURE 2.1
CUMULATIVE POPULATION VERSUS
DISTANCE FROM REACTOR
CONTAINMENT
(Midland Population Includes
Dow Personnel Onsite)
CUMULATIVE POPULATION (THOUSANDS)
DISTANCE (MILES)
BURLINGTON
ZION
INDIAN POINT
MIDLAND (TOTAL)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
150
160
170
2
3
4
5
6
7
8
9
10

providing features to mitigate accident consequences. The minimum radius of the exclusion area presently proposed is 500 meters. This will incorporate a small segment of land under the control of the Dow Chemical Company. This land consists of a fenced-in portion of the Dow waste treatment ponds in which no Dow employees are permanently located and which requires only occasional access by operating personnel. The applicant has stated that he will be cognizant at all times of the persons within that portion of the Dow Chemical property which falls within the exclusion area, and will exercise the right to remove any persons from the Dow property should conditions arise which warrant such removal.

The applicant has modified his initial proposal that the entire city of Midland be considered to be within the low population zone. Consumers Power Company now proposes a low population zone distance of one mile (1600 m). The residential population within the one-mile low population zone is 38. The business population within the low population zone, predominantly employees of Dow Chemical Company, is 1952. We find the Exclusion and Low Population Zone distances acceptable because (1) the residential population within the low population zone is very small, (2) the portion of the city of Midland within one mile of the facility consists almost entirely of the Dow Chemical Company complex and (3) Dow is equipped with a well-structured evacuation plan. The evacuation plan is discussed in Section 11.0.

2.2 Meteorology

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

Measured meteorological data are available from two wind stations at Dow Chemical Company which are located about 1-1/2 miles to the northwest of the site, and from the Saginaw Tri-City Airport, about eight miles to the southeast. The applicant has based his proposed diffusion model on the weather data taken at the Tri-City Airport as correlated with measured data from the Dow facility. This model is less conservative than the standard diffusion model we use.

We do not consider that the method used by the applicant to analyze these data is sufficiently precise to define atmospheric diffusion conditions, because the hourly weather data characterization technique may not accurately reflect actual conditions and the data are in the form of gross frequency distributions with no joint frequency distributions between stability, wind speed, and wind direction. We have concluded that (1) the available meteorological information presented by the applicant does not justify a departure from the standard model we use to determine the 2-hour and 30-day diffusion characteristics, and (2) the standard model we use (Pasquill Type F, 1 m/sec) provides a conservative basis for accident evaluations in the absence of adequate local meteorological data.

In calculating the doses at the nearest boundary of the exclusion area (500 meters) we have assumed a dilution factor of 6.7×10^{-4} seconds per cubic meter (Pasquill Type F, 1 m/sec) as opposed to the value of 2×10^{-4} seconds per cubic meter proposed by the applicant. Our consultant, Air Resources Laboratory, ESSA, concurs in our assumptions. Copies of the ESSA reports have been transmitted to the Committee.

The applicant has agreed to conduct an onsite meteorological program. We have informed the applicant that the scope of this program should be sufficient to provide a basis for the meteorological models he proposes to use for accident evaluation and routine release limits, including consideration of the effect of the cooling pond.

2.3 Geology

Our review of the site geology is not yet complete, and we will provide a supplemental report on this subject.

The principal item still under review arises because Dow has conducted brine removal at depths of 5000 ft and salt removal from the Devonian Detroit River group in the area at depths of 4100 ft to 4300 feet. Dow has selectively positioned the salt wells and recharged the brine deposits with depleted brine to minimize the potential for salt cavity collapse. We are concerned that the removal of salt from beneath the site might cause subsidence and differential settlement at the Midland Plant resulting in structural damage. We are engaging consultants to assist us in evaluating the applicant's assessment of the severity and consequences of potential subsidence. We will report to the Committee on this problem when our evaluation is complete.

2.4 Soil Mechanics

Our review of site soil mechanics is not yet completed. The applicant has stated that a factor of safety for slope stability of 1.1 is adequate for the design basis earthquake. Since no static or dynamic analyses have been performed for the plant fill slopes, we cannot evaluate the accuracy of the applicant's anticipated factor of safety. We are continuing our discussions in this area and will report to the Committee in a supplemental report.

2.5 Seismology

The seismicity of the site has been evaluated by the U. S. Coast and Geodetic Survey (USC&GS). Based on the review of the seismic history of the site and of the related geologic considerations, the USC&GS concludes that the applicant's proposal to use accelerations of 0.06g and 0.12g for the Operational Basis Earthquake and the Design Basis Earthquake, respectively, is acceptable. Copies of the USC&GS report will be transmitted to the Committee prior to the ACRS meeting.

The applicant has agreed to install a strong motion accelerograph. A description of the actual system, its location, and operating criteria will be provided at the operating license stage of our review. We find this to be acceptable.

2.6 Flooding and Hydrology

Our reviews of the hydrological aspects of the site and of the flooding potential of the Tittabawassee River are still continuing. We have requested information on the data and method of analysis used by the applicant to calculate the probable maximum flood level at the Midland site on three separate occasions (September 1969, January 1970, and February 1970). We are still awaiting sufficient information upon which to base our evaluation.

2.7 Environmental Considerations

The Fish and Wildlife Service has reviewed the application with regard to the consequences of release of radioactive waste material and heated waters to the environs. They have recommended that both pre- and post-operational ecological surveys, planned in cooperation with the appropriate federal and state agencies, be conducted. Their comments have been transmitted to the Committee and to the applicant. We will urge that the applicant comply with those portions of the Fish and Wildlife recommendations which are concerned with the release of heated water to the environment and will take into account those recommendations with respect to the release of radioactive materials to the environment.

A preoperational environmental radiation survey will be conducted at the Midland Plant site by the applicant. This program will consist of the following sample collections: six air particulate samples weekly, six air iodine measurements weekly, three gross beta measurements in waters of the Tittabawassee and Chippewa Rivers monthly, three tritium measurements on waters of the Tittabawassee and Chippewa Rivers monthly, and nine gamma scans on fish and other aquatic life monthly when possible. Sampling locations have been identified; eight of the eleven stations anticipated are located in an inner ring which will be within the influence of the plant when it is operating. Three additional stations are located approximately 20 miles from the facility to serve as background. We have evaluated the Midland Plant preoperational program and conclude that it will provide a valid basis for comparing future levels with preoperational levels.

2.8 Accidents at Dow Chemical Company

At our request, the applicant has evaluated the effect of accidents occurring at Dow on the Midland Plant. All Dow facilities presenting a serious explosive hazard are located at least one mile from the plant. The applicant has stated that none of the potential accidents will cause measurable damage at distances greater than 1,000 ft from the process unit involved.

In addition to the explosive hazard, large quantities of toxic chemicals are stored at the Dow complex. The failure of Dow storage tanks could result in significant concentrations of toxic chemicals at the Midland reactors. The applicant has stated that Dow has identified chlorine, bromine, and methyl bromide as having the maximum toxicity at the Midland Plant as a result of a release having a single initiating event. Assuming our standard meteorological model for accident analyses and the worst failure of a storage tank initiated by a single event, the applicant has calculated that the chlorine concentration at Midland would be 770 ppm. This concentration would be reached 20 minutes following the failure. On the other hand, if the chlorine releases were to occur with a 5 m/sec wind speed and Pasquill Type F stability exists, chlorine concentrations of 154 ppm would exist at the Midland Plant in four minutes.

The place the significance of these chlorine concentrations in perspective, the physiological effects of chlorine at various concentrations, as stated in "Dangerous Properties of Industrial Chemicals" (3rd Edition) by N. I. Sax, are presented below:

Limit for 8-hour continuous exposure	1 ppm
Detectable odor	3.5 ppm
Immediate throat irritation	15 ppm
Dangerous for short-term exposures	50 ppm
Fatal with brief exposures	1,000 ppm

The Consumers position with regard to this problem is that the Midland Plant will be connected to the office of the Dow Plant Protection Dispatcher by both telephone and short-wave radio. The plant would be notified within 2 minutes of any emergency at Dow and canister-type respirators or Scott Air-Paks could be donned to prevent operator injury.

These provisions alone do not give adequate assurance that personnel will be able to function with full effectiveness in the event of such an incident. We will require, therefore, that the control room and its ventilation system be designed to limit the concentration of toxic chemicals to less than that value which produces observable physiological effects (e.g., 15 ppm for chlorine).

3.0 PLANT DESCRIPTION

3.1 Nuclear Steam Supply System

3.1.1 Core

The nuclear steam supply systems provided for the Midland units will be two-loop Babcock & Wilcox Company pressurized water reactors. They will operate at a core power level 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. Engineered safety features and waste handling equipment have been sized based upon the 2552 Mwt ultimate power level. In addition, all accidents which result in releases of radioactivity have been analyzed based on the ultimate core power level. The power level of the Midland units is the same as that previously approved for Rancho Seco Unit I, Oconee Nuclear Station, Units 1, 2 and 3, and for Three-Mile Island Units 1 and 2.

The control rod holes in 16 of the fuel assemblies not equipped with control rod assemblies will be utilized as locations for fixed burnable poison rods. These are scattered symmetrically throughout the core. They ensure a moderator temperature coefficient of reactivity having a zero or negative value throughout the life of the core. Eight part-length control rod assemblies are provided for xenon control. The applicant has performed a modal analysis of xenon stability and studies are continuing in various geometries with codes which have thermal-nuclear iteration capability for both fuel and moderator temperature f.edback. Results to date indicate that (1) the core is stable in the radial direction, (2) the core design will not be susceptible to diverging azimuthal oscillations, and (3) potential axial oscillations can be controlled by the part-length control rods. We have evaluated the research and development effort planned to obtain more detailed information on the potential for xenon oscillations. We conclude that the proposed program is sufficient to develop a control scheme which will permit control of oscillations and that it can be completed by the time the operating license is issued.

3.1.2 Reactor Vessel Surveillance

The estimated end-of-life fluence for the reactor vessels is 3.0×10^{19} nvt, based on a 40-year service lifetime and a load factor of 0.80. Babcock & Wilcox states that the above value is conservative by 25%. B&W has checked their calculational model, the NRN Code, through various nuclear experiments. The experiments are outlined in answer to question 4.12 in Amendment 5 and indicate the code yields values of the fluence which either agree with experimental data or are conservative.

Consumers Power is participating in the Babcock & Wilcox Integrated Surveillance Program. Presently, B&W is in the process of rewriting the Topical Report which was originally submitted in July 1969 as part of the Duke Power Co. Oconee application. The report will not be issued in time for our review prior to the Midland ACRS meeting; however, on January 20, 1970, we met with B&W for a presentation of the revised

Integrated Surveillance Program. Based on this meeting, we have only some minor reservations, which we discussed with B&W. They are adopting changes to alleviate our concerns and thus we conclude that the proposed program will adequately monitor the change in transition temperature for the reactor vessels.

3.1.3 Reactor Internals

For normal design loads of mechanical, hydraulic and thermal origin, including the operational basis earthquake and anticipated transient loads the reactor internals will be designed to function within the stress limit criteria of Article 4, Section III of the ASME Boiler and Pressure Vessel Code.

All internal components are considered as Class I for seismic design and will be designed to withstand the loads which would result from the combined hypothetical earthquake and a loss-of-coolant accident. The strain limits for the internals material (304 SS) 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 welded by operators qualified and 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.

3.1.4 Reactor Coolant System

The reactor coolant system will be designed as a Class I (seismic) system to withstand the normal loads of mechanical, hydraulic and thermal origin plus anticipated transients and the operational earthquake within the stress limits of the appropriate codes given below.

The Midland Units 1 and 2 reactor vessels will be designed and fabricated in accordance with the 1968 edition of the ASME Code Section III, Class A plus the Summer, 1968 addendum, and Code Cases 1332, 1335, and 1339. The vessel design is the same as those intended for Arkansas Nuclear One, Crystal River 3 and 4, Rancho Seco 1, and the Three-Mile Island Plants. We have reviewed the planning for design and fabrication of the reactor vessels and the quality levels specified for these vessels. We conclude that the reactor vessels, as planned, should have an acceptable quality.

Reactor coolant piping will be designed to the USAS B31.7 Nuclear Power Piping Code dated February 1968 and including the June 1968 errata. In addition, all system components will be designed to withstand the concurrent blowdown and design basis earthquake loads.

The 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 inpingement and turbulent flow as well as natural frequency calculations which establish that a factor of at least two exists between possible resonant frequencies and known excitation frequencies such as pump blade passing frequencies and vortex shedding frequencies.

The application states that no vibration testing is planned for the Midland Plant. The basis for this position is the anticipated similarities between Midland and other B&W plants, as yet unspecified, which will be subject to vibration testing. The applicant has stated orally that limited measurements will be made during startup. We have informed the applicant that confirmatory vibration testing will be required as part of the Midland preoperational test program and that provision for such testing should be part of the design effort.

3.1.5 Pipe Whip

All Class I systems (including containment and ECCS) will be protected against pipe whip by (1) physical separation from Class ~~2 hig~~h pressure systems, (2) separation of redundant systems and/or components, or (3) restraint of lines which could whip and damage other Class I systems. We find these criteria acceptable.

3.1.6 Class I Mechanical Equipment

We have reviewed the analytical methods which the applicant proposes to use as the basis for the purchase specification to be issued for Class I mechanical equipment. For flexible equipment, the response spectra at the points of mounting would be determined from the time history of the component. For rigid equipment, the peak acceleration from the structural response spectrum at the level of support would be used. In addition, the quality assurance program calls for verification by the applicant of the analytical and empirical methods used by the vendors to certify that this equipment meets the specifications developed on the above bases. We find this approach acceptable.

3.2 Missile Protection and Flywheel Integrity

The applicant has proposed to protect the primary system and engineered safety features from missiles either by separation of redundant systems or by the use of missile shields. The applicant further states that in his design he will consider the orientation of components which could generate missiles and provide direct shielding to prevent missile generation from pressurized components from damaging other equipment.

Although the design has not progressed far enough to determine potential missile sizes and masses, the missile penetration formulae the applicant will use are acceptable to us. We conclude that the proposed missile protection planning is adequate. At the operating license review, we will evaluate the final design.

The primary pump-motor flywheels proposed for the Midland units will be fabricated from a modified A-516 material with a minimum specified yield strength of 45,000 psi. The rotational speed is 1200 rpm, the same as in Westinghouse plants, the OD is 65 inches versus 75 inches for a typical Westinghouse flywheel, the 12 inch thickness is the same as for Westinghouse designs. By our calculations, the bursting speed is 4050 rpm, thus the factor of safety against bursting is 3.38, which is about the same as provided at Ginna and Sequoyah. The applicant considered potential modes of bearing failure, and seismic loadings in his flywheel integrity analysis. On the basis of previous evaluations of similar flywheels, we conclude the flywheel design is acceptable in view of the intended high grade material, extensive quality assurance program, good manufacturing procedures and preservice surveillance requirements.

3.3 Inservice Inspection

The applicant has not yet developed his detailed inservice inspection plan, but states he will comply with the October 1969, Draft ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems. This version is very similar to ASME Section XI, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems, to be effective in April 1970. We conclude that an inspection plan developed in accordance with ASME Section XI is acceptable. In addition, Consumers is aware of our additional requirements for inservice inspection of the primary pump-motor flywheels, all primary vessel supports, and analogous inspections for the engineered safety features outside the primary coolant boundary.

3.4 Leakage Detection

Three means will be available to detect leakage from the primary system or other systems within the containment: (1) Four humidity detectors will be provided, one in each steam generator compartment, one at the refueling floor near the reactor refueling cavity, and one at the

600-foot elevation. They will be capable of detecting a change in humidity of 10%. (2) The reactor building sump level will generate an alarm when it reaches the three-inch step. This would correspond to approximately 250 gallons of primary water leakage. (3) Both radiation monitors within the containment which monitor the discharge of the reactor building air coolers are capable of detecting a 100 cc per minute leak in 45 minutes based on the primary coolant activity the applicant has calculated for operation with 1% failed fuel. The time necessary to detect leakage with this system will vary directly with leakage rate and inversely with primary coolant system activity. The Midland Plant array of instrumentation is sensitive, provides timely alarms, is redundant, and diverse. We conclude that the proposed leakage detection systems are acceptable. The applicant has not proposed a limit on the leak rate for plant operation. This value will be established during preparation of technical specifications for the operating license.

3.5 Sharing

The following systems are shared between Midland Units 1 and 2: (1) The spent fuel storage pool and the storage pool cooling system, (2) the service water system, (3) the auxiliary building heating and ventilation systems, (4) the fire protection system, (5) the process water makeup system, (6) the cooling water pond, (7) the water treatment plant, (8) the auxiliary building and other ancillary buildings, such as the administration building, machine shops, laboratories, etc.,

(9) the radioactive waste treatment system, (10) borated water storage tank, (11) the purification demineralizer, and (12) condensate storage tank. With the exception of the service water system and the borated water storage tank, none of these systems are required to achieve normal shutdown or are required under accident conditions.

The service water system consists of two major loops servicing vital equipment. Redundant equipment is placed on both loops. A failure in either loop does not affect the other. Therefore, each of the two loops provides the minimum engineered safety feature requirements for both units.

The borated water storage tank has a capacity of 680,000 gallons. This capacity is based on the requirement that after filling one unit's containment and auxiliary building refueling canal to a depth of 23 feet above the reactor vessel flange base (approximately 400,000 gallons) 250,000 gallons of borated water would remain for loss-of-coolant accident protection for the operating unit plus an additional 30,000 gallons for margin. The accident analyses are based on 250,000 gallons of water available for the emergency core cooling system and the reactor building spray pumps.

Based on the foregoing, we have determined that the sharing proposed has no adverse effect on plant safety.

3.6 Cooling Pond

The ultimate heat sink for the facility is the cooling pond which surrounds the plant. The pond is twelve feet deep and has a surface area of 880 acres when full. The total storage provided is 12,600 acre-feet. Of this, 7900 acre-feet are usable for heat dissipation. This provides capacity to satisfy plant cooling needs for 100 days without drawing water from the Tittabawassee River. During the summer, the service water system heat load will be rejected in two mechanical-draft cooling towers, with makeup drawn from the cooling pond.

To provide a source of emergency cooling water in the event the Class II cooling pond dikes should fail as the result of an earthquake, a 24-acre area in the northeastern corner of the cooling pond has been excavated to a depth of 6 feet below surrounding grade. The intake invert extends to 2 ft above the bottom of the emergency pond, and this reservoir has a useful capacity of 70 acre-feet. If water seepage into the soil is considered, this capacity is sufficient to reject plant decay heat for 30 days without makeup.

Two supply and two return lines will draw service water from the emergency cooling pond and return it to the pond. The intake lines will include bar screens to prevent debris from entering the service water system. It will be necessary for the applicant to monitor for silting in the emergency pond and if necessary, to dredge it periodically. Further, to assure that the service water intakes are not damaged by the debris from a flood, we will require that (1) the dikes be capable of withstanding the consequences of the Probable Maximum Flood without breeching or overtopping, or (2) the bar screens be designed to cope with the debris which might enter the pond under flood conditions.

3.7 Structures

3.7.1 Class I Structures Other Than Containment

The applicant has defined Class I structures, systems and equipment as those whose failure could cause release of radioactivity which would exceed 10 CFR 20 limits at the site boundary, or those necessary for safe shutdown. We concur in this definition and have examined the applicant's categorization of plant structures and components to assure that this criterion has been followed. We find the applicant's categorization acceptable.

The applicant has considered the interaction of Class I and Class II components and structures. Even though the turbine building is not considered a Class I structure, it is designed in such a manner that it will not collapse under seismic or tornado loadings.

The applicant has stated that tornado-protected structures will be designed to withstand a tornado with 300 mph rotational velocity, 60 mph translational velocity, and a pressure drop of 3 psi in 3 seconds. A uniform 300 mph wind front will be assumed vertically and horizontally across the reactor building, but a 360 mph peak wind velocity will be used on smaller Class I structures which must be designed for tornado loads and for which the assumption of a velocity gradient would not be conservative. Design stress limits will be 90% of the yield stress of the reinforcing and 75% of the concrete ultimate compressive stress. We and our design consultants agree with this approach.

3.7.2 Containment Structures

3.7.2.1 Description

The containment structure is similar to other Bechtel designed PWR containments (Rancho Seco, Russelville) in that it is a prestressed concrete cylinder and dome supported on a reinforced concrete foundation slab. It also will use the large size BBRV tendon system (approximately 170 wires).

3.7.2.2 Containment Functional Evaluation

We have investigated the pressure response of the containment to the loss-of-coolant accident. Various loss-of-coolant containment pressure transients were investigated by the architect-engineer, Bechtel Corporation, using their COPATTA computer code. We compared the COPATTA code with the CONTEMPT code, developed by INC, which we have been using as an independent means of computing pressure transients.

The comparison between the codes showed two significant differences. One is in the equations that determine how the primary coolant water separates into steam and saturated water during the blowdown process. Use of the COPATTA equations results in less energy in the steam, with a resultant lower containment pressure than that predicted by the CONTEMPT code. The second significant difference between the two codes is in the value used for the structural condensing heat transfer coefficient. The COPATTA code uses a correlation based on the work of Tagami[1] while

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[1] Tagami, T. "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)."

CONTEMPT uses a condensing heat transfer coefficient based on the work of Uchida.[2] In general, lower peak containment pressures will be obtained when the Tagami data are used.

After several meetings with the applicant and Bechtel, we concluded that:

1. The use of the thermodynamic equations in the COPATTA code results in a nonconservative prediction of the containment pressure transient and is not acceptable.
2. The use of the Tagami data instead of the Uchida data is acceptable. The Tagami data were obtained for transient conditions that more closely resemble accident conditions than the steady-state data of Uchida.

In reaching these conclusions, we had the benefit of work done by the DRL Technical Assistance Program at the Idaho Nuclear Corporation (INC). In particular, a review of the "state of the art" of film condensing heat transfer was made by INC and a report on this subject will be issued shortly. Our conclusions on the COPATTA code are shared by the people at INC.

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[2] Uchida, H., Oyama, A, and Togo, Y. "Evaluation of Post-Incident Cooling, Systems of Light Water Power Reactors," Third International Conference on the Peaceful Uses of Atomic Energy, New York, 1965.

We performed calculations using the CONTEMPT code for the blowdown mass rates and enthalpies supplied by the applicant, and for the 14.1 ft^2 break area only. The following table compares the applicant's COPATTA calculations with our CONTEMPT calculations:

TABLE I

	Peak Containment	Pressure (psig)	
Break Area,	COPATTA	CONTEMPT*	CONTEMPT**
14.1	53.0	57.5	55.6
8.5	54.0		
5.0	54.1		56.8***
3.0	53.0		
1.0	49.2		

*Using Uchida data

**Using Tagami data

***This pressure was estimated as follows:

Based on these comparisons, there is a 1.9 psi pressure difference due to the use of Tagami versus Uchida data. There is another 2.6 psi pressure difference associated with the two thermodynamic models. The CONTEMPT runs were made for the 14.1 ft^2 break area and highest pressure calculated by the COPATTA code was for the 5.0 $foot^2$ break area. The Midland peak pressure is 56.8 psig using the Tagami data, the CONTEMPT thermodynamic model, and the 5 $foot^2$ break area.

As a result of our evaluation, the applicant has increased the containment design pressure from 58 psig to 62.5 psig. This provides a ten percent margin over the peak calculated pressure, which we consider acceptable.

3.7.2.3 Containment Structural Evaluation

The containment liner will be welded 1/4-inch thick 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 percent. The concrete will utilize Type II cement and will have 28 day compressive strengths of 5000 psi in the containment walls and dome, and 4000 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. These splices will be performed in accordance with strength and testing criteria that we have found acceptable. The proposed prestressing system is the same as that we reviewed and found acceptable on Arkansas Nuclear One, and Consumers Power has agreed to furnish us with essentially the same detailed proof-testing and system evaluation data on which approval of the Arkansas Nuclear One prestressing system is to be based. Therefore, we have concluded that these materials are acceptable for the containment structures.

The design loads and load combinations under normal operation, and under accident, seismic, and tornado conditions have been evaluated by us and our design consultants and found to be acceptable.

The applicant has not fully demonstrated the applicability to this site of the Housner seismic spectra which are proposed as the basis for the dynamic design of certain Class I structures, piping, and equipment. We and our seismic design consultants have concluded that, with the justification of these spectra, or the use of more conservative spectra, the applicant's seismic design approach would be acceptable.

Response spectra are also developed at different locations in the structure in order that Babcock & Wilcox and other vendors can then be informed of the seismic loads which the equipment they furnish must be capable of withstanding. When equipment is located on a flexible floor, amplification in the vertical direction of ground level acceleration is also included.

We and our design consultants find the applicant's criteria for the dynamic design of Class I piping and equipment to be acceptable.

3.7.2.4 Testing and Surveillance

An initial structural proof test at 71.9 psig (115% of containment design pressure) and an initial leak rate test at 62.5 psig and several lesser levels will be performed. A periodic leakage rate test at 25 psig will be performed at a frequency to be established in the technical specifications. Periodic structural surveillance will be in the form of lift-off readings on nine representative tendons, and three wires of a tendon in each of the three directional groups (hoop, vertical, dome) will be removed and inspected. We conclude that both the pre- and post-operational testing programs are acceptable for the construction permit review. Bechtel is, however, developing a statistical study which will describe in detail the Palisades tendon surveillance program on which the Midland program is based. We will review this study and its conclusions, and if a change in Midland's tendon surveillance program is then indicated, we will require it at the time of the operating license review for the Midland Plant.

3.8 Instrumentation and Control

The instrumentation and control systems have been evaluated against the Commission's Proposed General Design Criteria (GDC), 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 28, 1968. The reactor protection instrumentation and control systems as well as the instrumentation which initiates and controls the engineered safety features are substantially the same as those proposed and found acceptable in the Three Mile Island Nuclear Station, Unit No. 2 (TMI-2) design. The following discussion is limited to those features of the design which differ from TMI-2 and to those for which new information is available. These areas concern only the engineered safety feature (ESF) instrumentation design and the requirement for a diverse engineered safety feature initiation signal.

In the TMI-2 design, three instrument channels are provided to monitor each variable required to initiate ESF. These instrument channels are arranged in a two-out-of-three (2/3) coincidence logic for initiation of the protective action. The Midland design uses four instrument channels arranged in two-out-of-four (2/4) coincidence logic to initiate a protective action. The applicant has stated that the systems will meet the requirements of IEEE 279. We have concluded that this modification 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 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 since adequate redundancy is provided and the system is designed to IEEE 279.

In the Midland design, the safety injection system is actuated by either low reactor pressure or high containment pressure signals. However, reactor trip is not initiated by a high-containment-pressure signal. A number of the analyses on which the effectiveness of the ECCS are based take credit for a reactor trip. The applicant was requested to provide an analysis to show that in the event of a failure of the low reactor pressure signal the high containment pressure signal alone, or in coincidence with other signals which do not depend on low reactor pressure, will assure the effectiveness of the ECCS. The applicant responded by stating that because of the reliability of the low reactor pressure signal, the high containment pressure signal for actuation of ECCS is unnecessary. He proposes, however, to conduct a study to determine whether diverse ECCS signals are required. This will be included in the forthcoming B&W evaluation of common mode failures. Further, the commitment was made to add whatever forms of protection are found necessary by this study. We have informed the applicant that we plan to require diverse ECCS signals, and that either signal provide a reactor trip to assure the effectiveness of the core cooling system. The applicant has not yet agreed. We will require that diversity be provided.

3.9 Electric Power Systems

3.9.1 Offsite Power

The Midland Plant, Units 1 & 2, will be interconnected to the Consumers Power Company's system through 345 kV and 138 kV circuits. Power from each unit's generator is fed via separated circuits to the 345 kV switchyard. The 345 kV switchyard is interconnected to the adjacent 138 kV switchyard by means of stepdown transformers. Both switchyards are 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 fault in the electrical power system. The design criteria include the requirement to maintain system stability with the sudden outage of all generating capacity at any generating plant. We conclude that if the reliability requirements are properly implemented in the design, the loss of the Midland Plant should not cause interruption of offsite power.

Two startup transformers provide redundant, independent sources of offsite power to the 4160 volt engineered safety feature buses of Units 1 and 2. One startup transformer is supplied by a 138 kV transmission circuit from the 138 kV switchyard. This circuit is mounted on independent towers and shares the same right-of-way with the two 345 kV circuits connecting each unit's generator output to the 345 kV switchyard. These circuits, however, are separated sufficiently along this common right-of-way to preclude the loss of adjacent circuits should one tower fall.

The second startup transformer is supplied by a 138 kV transmission circuit connected to the Dow South Substation. This circuit is mounted on independent towers and on a right-of-way separated from that of the first startup transformer. It is our understanding that this substation is fed from multiple 138 kV system transmission circuits and that the controls are under the direct supervision of the reactor operator. Upon loss of the normal supply, each transformer is automatically connected to one of two engineered safety feature buses in each unit. Therefore, the loss of one startup transformer will result in the loss of offsite power to one of two redundant engineered safety feature buses in both units.

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

3.9.2 Onsite Power

The design of the onsite 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 carry minimum safety feature loads in one unit and minimum safe shutdown loads in the other unit without exceeding the continuous rating. The applicant's preliminary load calculations indicate that a diesel generator with a 3000 kW continuous rating is required. The applicant has agreed to supply test data to confirm the suitability of this large capacity machine 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 building so that an incident in one diesel generator or bus will not involve its redundant counterpart either physically or electrically. Each diesel generator will be provided with a day tank of sufficient fuel capacity to permit four full-power-hours of operation. The main fuel storage facility will have sufficient capacity to assure the operation of a diesel generator fully loaded, for seven days.

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

The preferred 120 volt a-c system for the protection instrumentation systems 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 d-c buses. This arrangement provides four independent power sources for the protection system instrumentation of each unit.

We have concluded that the design of the onsite power system is acceptable.

3.9.3 Cable Design, Selection, Routing, and Identification

The applicant has documented his criteria for cable design, selection, and routing. These have taken into consideration the loss of redundant channels of protection from a single cause such as fire, and the identification of safety related circuits and components. We find the applicant's criteria acceptable.

3.9.4 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 these 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 engineered safety features, and emergency electric power systems. These requirements will be satisfied by analysis or by providing applicable test results.

We conclude that the applicant's environmental testing program is acceptable.

4.0 ENGINEERED SAFETY FEATURES

4.1 Emergency Core Cooling System

The emergency core cooling system (ECCS) is the same as that proposed for the Three Mile Island 1 & 2, Crystal River, Rancho Seco, and Arkansas Nuclear One plants. Single failures of active ECCS components and single passive failures during long-term cooling can be tolerated without jeopardizing plant safety. All piping for the ECCS will be designed in accordance with the USASI Code B31.7.

The ECCS for the Midland Units will consist of the following systems:

(1) High pressure injection system - This system normally operates as part of the makeup and purification system. Two independent and redundant systems with three high pressure pumps each inject a minimum of 340 gpm into the primary inlet piping. The system is initiated by a low primary system pressure signal ($\leq$1500 psig) or a high containment pressure signal.

(2) Core flooding system - There are two independent and redundant storage tanks containing a total of 1880 cubic feet of borated water. This water is discharged automatically via two separate 14-inch diameter nozzles into the reactor pressure vessel when primary pressure system drops below approximately 600 psi.

(3) Low pressure injection system - This system normally operates for shutdown cooling as part of the decay heat removal system.

It consists of two independent and redundant systems. Each system provides 3,000 gpm flow into the reactor vessel at a vessel pressure of 100 psi. The system discharges through the same 14-inch diameter nozzle used for injection of core flooding tank coolant into the pressure vessel when primary system pressure drops to 200 psig. The system is initiated by a low reactor coolant system pressure initiation signal (<200 psi) or a high reactor building pressure signal.

The source of coolant for both the ECCS high pressure injection and low pressure injection is a 650,000 gallon borated water storage tank. All emergency injection coolant will be maintained at the minimum concentration of 2270 ppm of boron. The pressure and level of these tanks will be displayed in the control room, and alarms will sound when any condition is outside the normal limits. The water will be periodically sampled and analyzed to assure proper boron concentration.

The ECCS design is based on limiting the maximum fuel clad temperature to less than 2300°F for any size primary system pipe rupture up to the double-ended rupture of the 36-inch diameter outlet pipe. In analyzing the core thermal transient following the loss-of-coolant accident, the assumption is made that core flooding tanks, one high pressure injection pump, and one low pressure injection pump provide coolant to the core. Delivery of the coolant at the initial flow rate of 3500 gpm is assumed not to start until the primary system has depressurized to 100 psi or after 25 seconds, whichever occurs later. The maximum fuel clad temperatures for a spectrum of hot leg and a cold leg break sizes are shown in Table 4.1.1. The maximum temperatures shown assume the reactor has operated at a power level of 2552 megawatts thermal. The method of analysis is the same as has been used on previously reviewed B&W reactors.

We have concluded that the design of the proposed ECCS (1) limits the peak clad temperature to well below the clad melting temperature, (2) limits the fuel clad-water reaction to less than 1% of total clad mass, (3) terminates the temperature transient before the clad geometry necessary for core cooling is lost and before the clad is so embrittled as to fail upon quenching, and (4) reduces the core temperature and removes core heat until the core will remain covered without recirculation and replenishment of coolant. The ECCS is designed to provide this protection for all sizes and location of pipe breaks up to and including the instantaneous double-ended rupture of the largest reactor coolant pipe. As indicated in Section 3.8, we will require that diverse actuation signals be provided, either one of which will assure the effectiveness of the emergency core cooling system.

TABLE 4.1.1 PEAK CLAD TEMPERATURE FOR LOCA

Rupture Size (ft^2)	Rupture Diameter (in)	Hot Spot Maximum Temperature (F)	
		Hot Leg Ruptures	Cold Leg Ruptures
14.1	36	2007	--
8.5	28	1893	1694
5.0	21.4	1584	1802
3.0	16.6	1328	1527
1.0	9.6	1094	1309
0.4	6.1	1098	1105
0.0575	3.25	1080	--

4.2 Cavity Design and Cavity Flooding

Provisions will be incorporated in the design to allow for the addition of a system which will provide for rapid and continued flooding of the reactor vessel cavity up to at least the reactor coolant piping nozzles. The water will then overflow and return to the reactor building sump. As presently envisioned, the applicant has stated that such a system would consist of the following:

1. Reactor cavity flooding tanks and associated piping and valves located within the reactor building will fill the reactor up to the overflow point. Actuation of this system would be by a combination of core-flooding tank low-level and a low pressure safety injection initiation signal.

2. A gravity drain line from the lower end of the reactor building refueling canal which would allow the water from the building sprays, and upper building floor drains, which is collected within the refueling canal, to be routed to the reactor cavity.

The cavity is designed for a differential pressure of 250 psi. This pressure would be reached in the event of a 3 square foot slot break in the reactor coolant system piping.

4.3 Iodine Removal

The proposed iodine removal spray system for Midland is similar to that of the Three Mile Island Units 1 and 2 and Arkansas Nuclear One. The Midland design will employ alkaline thiosulfate solution mixed into the two independent 1300 gpm containment spray systems to increase the iodine removal capability following the loss-of-coolant accident. Sodium thiosulfate and sodium hydroxide are added to the system by separate redundant metering pumps.

We have informed B&W and the applicant that the long term stability of the alkaline sodium thiosulfate solution under post-DBA conditions has not been demonstrated to our satisfaction. B&W is continuing the current R&D program to study this aspect of the problem.

The materials compatability aspects of the spray solution with all exposed construction materials have not been completely evaluated by the applicant, and further research and development effort is required.

During the initial spray phase when spray water is drawn from the storage tanks, the spray solution as it enters the containment is always alkaline but will never exceed a PH of 11. After mixing is complete, the initial composition of the mixture of spray water, ECCS water, and reactor coolant system water will have a pH of approximately 9. The applicant has stated that the 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 model than that used by the applicant, yielding a spray removal constant (λ) of 2.5 hr^{-1}. The fundamental model for the evaluation of iodine removal by containment sprays was developed by Griffiths [AERE Report AHSB (S) R45 (1963)]. We have used conservative values for all parameters, allowing for minimum system performance and maximum uncertainties.

We have informed the applicant that space should be reserved for charcoal filters should the containment spray R&D program fail to meet its objectives.

4.4 Containment Heat Removal Systems

The containment heat removal systems consist of a spray system and a fan cooler system. The spray system is designed to deliver 2350 gpm through the spray nozzles within 35 seconds after the initiation of the loss-of-coolant accident based on the diesel generators loading sequence with both pumps operating. We have evaluated the capability of this system and agree with the applicant's analysis that the system is capable of removing 200 x 10^6 Btu per hour from the containment atmosphere at design conditions with both spray pumps operating.

The containment building air cooling system consists of four separate units each operating continuously and independently of the others. The coolers are located at the lower level of the reactor building. Each unit consists of a roughing filter, a cooling coil section, and two fans of equal capacity operating in parallel. Following the loss-of-coolant accident, each cooler will continue to operate with one fan in service and will have a heat removal capacity of 50 x 10^6 Btu per hour at the following conditions: air-steam inlet temperature 286°F, cooling water flow 1200 gpm, cooling water inlet temperature 100°F. Following the accident, the only changes in cooler operation are an increase in service water flow by opening a stop valve, and the stopping of one fan on each cooler.

The cooling coils are similar to those tested for the Palisades Plant also owned by Consumers Power Company. The coils will be sized using the same techniques used to size the Palisades coils. We conclude that the cooling units are adequately designed.

5.0 QUALITY ASSURANCE

The applicant has described the Quality Assurance Program which he proposes to utilize for the design and construction phases of the Midland Plant Units 1 and 2 and has stated that the Quality Assurance Program is intended to satisfy the intent of proposed Appendix B to 10 CFR 50.

Consumers Power Company has contracted for the services of Bechtel to perform the architect-engineering and construction efforts and of Babcock & Wilcox to supply the NSSS. The applicant and both major contractors will prepare and utilize quality assurance programs which encompass each company's scope of effort, including their subcontractors. In addition, Bechtel will perform auditing of the quality assurance efforts of B&W and other contractors of the applicant. The extent of the applicant's participation in the overall quality assurance program for the project consists of a review of each principal contractor's quality assurance program, and audits and surveillance of the contractors to assure proper implementation of the program. We judge this degree of participation by the applicant to be sufficient. The quality assurance programs of the applicant and principal contractors will be performed in accordance with written policies, procedures, and instructions.

A Quality Assurance Engineer is provided within the applicant's organization who is directly responsible to the Manager of General Plant Engineering, as is the Project Engineer. The Quality Assurance

Engineer's function is to direct the applicant's quality assurance efforts in order to determine that the principal contractors' quality assurance programs are being properly executed. The applicant's organizational arrangement provides the necessary independence of quality assurance from the pressures of cost and schedule. The applicant, since he performs no design and construction efforts, can conduct the quality assurance audits on an adequately independent basis.

In his description of the overall quality assurance program for the project, the applicant has presented the approach he and his principal contractors will follow for each of the quality assurance criteria specified in proposed Appendix B to 10 CFR 50. His program requires that planned and documented actions be applied to all quality related activities which involve the structures, systems, and components important to safety. We conclude that the applicant's planned approach for each of the critical areas is satisfactory for the construction permit stage. During plant construction we will follow the development of the details of the applicant's quality assurance program.

6.0 DOW USAGE OF PROCESS STEAM

Approximately 75 percent of the steam heat energy supplied by the nuclear steam supply system will be used to generate electrical energy. Steam containing the remaining heat energy will be transported to the site boundary for process use by the Dow Chemical Company. Under normal conditions 400,000 pounds per hour of process steam at 675 psia will be throttled from the Unit No. 1 main steamline and exported to Dow Chemical and 3,650,000 pounds per hour of process steam at 197 psia will be removed from the Unit No. 1 turbine moisture separators and exported to Dow. Most of the steam is condensed and returned to the secondary system as feedwater. The steam not condensed is replaced by treated makeup from Dow. The main steam piping is arranged so that the Unit No. 1 main steam system can be supplied with steam from Unit No. 2 when the Unit No. 1 is out of service. The steam exported to Dow will be used as process steam in manufacture of industrial chemicals, agricultural chemicals, and pharmaceuticals.

The applicant will monitor secondary steam for radioactivity. Monitoring will be based upon the detection of the presence of nitrogen-13. This will be used as a tracer to determine if there is any leakage from the primary system to the secondary system.

Nitrogen-13 is produced by a (p, α) reaction with the oxygen-16 nucleus, the proton originating from a hydrogen recoil from fast neutron interaction. Nitrogen-13 decays with a positron emission and the detection system proposed will be based on the detection of the 0.5 Mev annihilation gamma. The continuous gamma monitor to be used at Midland will consist of a sodium-iodide crystal scintillation detector surrounded by a large volume of liquid sample. A level of 3×10^{-6} microcuries per milliliter of nitrogen-13 in the sample will produce a net count rate on the instrument roughly equivalent to background. The detection of nitrogen-13 activity will provide no direct information concerning the presence of other radionuclides in the process steam. It will provide an indication of the magnitude of primary-to-secondary leakage rate. This information, combined with an isotopic inventory of the primary system obtained through batch sampling, will permit the applicant to calculate the process steam activity inventory on an isotopic basis.

At the present time, we cannot determine if the monitoring scheme proposed has sufficient sensitivity to measure 3×10^{-6} μ Ci/ml. The applicant will undertake a research and development program to determine the ability of this system to detect the presence of nitrogen-13 in the steam phase. This program will be conducted at the Big Rock Point Nuclear Plant and at Palisades, and the applicant will determine the partition factor for nitrogen as the steam condenses, and the properties of the detector.

Although preliminary discussions have been held with Dow Chemical on the use of steam in the production of their products, such uses have not been considered in this report. Preliminary indications from data supplied by Dow show that for the level of detection proposed, the quantities of radioisotopes which could be transferred to the products would be extremely small. A decision as to how the Commission will exercise control over distribution of steam will depend upon further review of its intended uses and controls proposed. Nevertheless, Consumers will be required to operate the plant such that effluents, including process steam, meet the requirements of the Commission's Regulations (10 CFR 20 and 10 CFR 50). On this basis, we have concluded that the export of process steam derived from the reactor secondary system beyond the plant boundary is acceptable, provided a suitable detection system is developed. If this R&D program fails to demonstrate that the proposed detection system is adequate, we will not permit the export of process steam until an acceptable system has been installed and tested.

Distribution of products may require approval of other Federal agencies. The Food and Drug Administration would have control of distribution of many of the Dow products. The proposal by Dow to use the steam has been discussed with FDA by both Dow and the AEC, but definitive agreement on conditions to be met for FDA approval has not been reached. The Delaney amendment to the FDA Act requires "...that no additive shall be deemed to be safe if it is found to induce cancer when ingested by man or animal, or if it is found, after tests which are appropriate for the evaluation of the safety of food additives, to induce cancer in man or animal." This clause had been interpreted by the courts to prohibit the use of any additives which under any conditions induce cancer in any strain of test animal.

7.0 ACCIDENT ANALYSES

7.1 Reactivity Transients

7.1.1 Uncontrolled Rod Motion

The applicant has analyzed two accidents which might be initiated by control rod motion, viz, rod withdrawal during startup, and rod withdrawal at rated power. A reactor short period trip was not incorporated in the analysis. Sensitivity analyses were performed in which trip delay time, reactivity addition rate, the moderator temperature coefficient of reactivity, and the Doppler coefficient were varied.

For the startup accident, the nominal 1.5% delta k/k rod group has a reactivity addition rate of 9.2×10^{-5} delta k/k per second. The peak thermal heat flux calculated by the applicant is 57% of the maximum full power heat flux, assuming the period trip is inoperable. A peak pressure of 2515 psia, the relief valve set point, is also reached. The capacity of the relief valves is adequate to handle the maximum rate of coolant expansion resulting from the startup accident.

A rod withdrawal accident at power was analyzed in similar fashion. The transient is terminated by a high neutron flux level trip at 114% full power and the calculated reactor thermal power is limited to 108%.

These analyses assume that the pressurizer level is at its normal operating condition. Since no pressurizer level trip is provided in the safety system design, the applicant has analyzed the various malfunctions necessary to cause the pressurizer level to rise above nominal value and above the high level alarm point. These analyses indicate that even if the pressurizer level controller fails during power operation, 35 minutes are required to fill the pressurizer with one charging pump operating. During this time, after 15 minutes a high pressurizer level alarm would be initiated and after 20 minutes the low letdown tank level alarm would be sounded.

The applicant has stated that good operating practice calls for a known pressurizer level to be established prior to performing any rod withdrawal operation when shut down. In addition, Consumers Power has stated that when the reactor is being prepared for startup, the written procedures will call for a minimum pressurizer level, rather than a normal or an above normal level. The applicant has stated that, if required, a technical specification restriction requiring that a given maximum pressurizer level be verified prior to pulling control rods could be imposed. Based on these considerations, we have concluded that the addition of a pressurizer high-water-level scram would not provide significant added protection and, therefore, is not required. However, in view of the importance of high-pressurizer-level alarm, we will require that the high-pressurizer-level alarm system be designed to reactor protection system standards.

7.1.2 Rod Ejection Accident

The applicant has analyzed the rod ejection accident for beginning-of-life and end-of-life conditions at both 2552 Mwt and zero power. The maximum worth of a single control rod at power with no xenon present is 0.46% delta k/k. The maximum worth of a single control rod at hot zero power critical conditions is 0.56% delta k/k. The applicant has analyzed the transients resulting from ejected rods of various worths using the KAPP-1 digital computer program. This code contains a two dimensional heat transfer model, and a point kinetics physics model. As a check on the KAPP-1 calculation, the rod ejection accident was also analyzed for a limited number of cases using the WIGL-2 digital computer program, a one-dimensional space-dependent kinetics code. The WIGL-2 calculations performed have been for the full power, beginning-of-life case.

The analyses performed by the applicant indicate that no DNB or fuel damage will result from a rod ejection accident at zero power critical. His analyses indicate that the peak fuel enthalpy for the hottest rod would be approximately 55 calories per gram.

For a rod ejection accident at 2552 Mwt, the point kinetics models predicts a peak fuel enthalpy of approximately 170 calories per gram, a peak thermal power of 126% at beginning-of-life, and 4.1% of the fuel rods experience DNB. (For comparison, the space-dependent kinetics model predicts a peak fuel enthalpy of 130 cal/gm)

We have estimated the potential offsite doses resulting from this accident, assuming (1) 4.1% of the fuel rods in the core perforate releasing 20% of the noble gases and 10% of the iodine in these rods to the primary system, (2) no leakage from the primary system, (3) the core is operating with the coolant activity associated with 1% failed fuel, (4) the 1 gpm primary-to-secondary system leakage in the steam generator at the time of the accident, varies as the square root of the delta P as the steam generator and primary system reduce in pressure, (5) loss of offsite power requiring heat rejection by boiloff to the atmosphere in the steam generators, and (6) boiloff in the steam generators results in the release of equilibrium secondary activity and the activity in the primary system leakage to the atmosphere with no partition factor for iodine. The resultant calculated doses are presented in Table 7.6.

As previously stated, the space-dependent and point-kinetics results have been compared for a limited number of rod worths for the beginning of life ultimate power level case. This comparison indicates that the space-time dependent solution yields a lower peak fuel enthalpy for rod worths of 0.5% delta k/k or less and, therefore, it is conservative to analyze the consequences of ejection of the maximum worth rod at power (0.46% $\Delta k/k$) using the point-kinetics method. Since the slope of a plot of peak fuel enthalpy versus rod worth is significantly higher for the WIGL-2 calculation than for the point kinetics calculation, we will require that at the operating license stage of our review, the applicant perform space-dependent kinetic calculations for both beginning-of-life and end-of-life conditions at both ultimate power and at zero power.

7.2 Accidents Resulting from Mechanical Failures

7.2.1 Steam Generator Tube Rupture

We have analyzed the results of a steam generator tube failure assuming a double-ended rupture of one steam generator tube occurs with unrestricted discharge from each end. The applicant has stated that this will trip the reactor on low pressure in about eight minutes. Isolation of the affected steam generator can be effected since the operator can identify the problem from (1) the low reactor coolant pressure, (2) the pressurizer level, and (3) the early increase in radioactivity in the steamline from the affected steam generator. Under these conditions it will require 15 minutes to cool the reactor system down to the temperature corresponding to the saturation pressure at which the atmospheric dump valve is set. Assuming the operator takes no action until the reactor trips, the total time required to isolate the secondary side of the affected steam generator is 23 minutes. The double-ended rupture of a steam generator tube is well within the capacity of the core cooling system. Thus, core damage is not assumed to occur.

We have calculated the offsite doses assuming (1) the tube rupture occurs concurrent with the loss of offsite power resulting in a loss of condenser flow, thus preventing the use of the condenser for a decontamination factor, (2) the operator does not isolate the affected steam generator, (3) 5650 ft^3 of the primary system coolant blows down to the steam generator and is released in two hours, (4) no iodine partitioning is available because portions of the steam generator tubes are uncovered, and (5) the plant has been operating with primary coolant activity corresponding to that calculated by the applicant for 1% perforated fuel rods assuming an average gamma energy of 0.7 Mev. Doses are presented in Table 7.6.

7.2.2 Steamline Rupture

The applicant has established the following criteria to govern the environmental effects of a steamline failure:

1. The reactor shall trip and remain subcritical until a controlled rate of system cooldown can be effected.
2. There will be no fuel damage as a result of the transient.
3. No steam generator tube damage will occur due to the loss of secondary side pressure and the resultant temperature gradients.
4. Doses will be within acceptable limits.

The applicant has analyzed the consequences of the double-ended 36-inch steamline rupture. In this analysis, an analog-hybrid computer program was used to study the transient characteristics of the reactor coolant system and the steam generator. This model included a detailed analog description of the secondary side of the steam generator, energy balances for the principal steam generator components, the entire reactor coolant system, and pressurizer, and reactor kinetics, trip logic, and fuel pin simulation with Doppler and moderator temperature coefficient feedback. These analyses indicate that the reactor will trip six seconds after failure and will not return to criticality thereafter. The maximum thermal power during the transient is 109%. Thus, no fuel damage is expected.

We have estimated the radiological consequences of a steamline rupture assuming (1) the reactor trip associated with the steamline failure results in a loss of offsite power and (2) the plant has been operating with one gpm primary-to-secondary leakage and that this leakage occurs in the steam generator not affected by the steamline rupture, and therefore, depressurization in the primary system must be accomplished by boiloff in the steam generators not affected by the steamline failure. If it is assumed that no fuel damage occurs, the doses are less than 1 rem. We are continuing to evaluate the applicant's code in connection with our review of the Duke Power Company Oconee application.

7.3 Refueling Accident

A refueling accident can result if a fuel assembly is dropped or otherwise damaged during transfer from the reactor vessel to the spent fuel storage pit.

The applicant has stated that the maximum damage that can occur would result in a release of gap activity from one row of fuel rods in a fuel assembly. We have evaluated the consequences of this accident assuming: (1) all 208 rods in one fuel assembly fail, releasing 10% of the halogens and 20% of the noble gases associated with that fuel element, (2) a reduction factor of ten as the iodines pass through the refueling water, (3) an auxiliary building charcoal filter efficiency of 90% for elemental iodine, and (4) 24 hours decay prior to refueling. The resultant calculated doses are presented in Table 7.6.

7.4 Gas Decay Tank Rupture

We have analyzed the radiological consequences of a gas tank rupture by assuming the tank contains a quantity of noble gases equivalent to that which we calculate would be present in the primary system when operating with 1% of the fuel rods experiencing clad defects. Our calculated doses are presented in Table 7.6.

7.5 Loss-of-Coolant Accident

7.5.1 Radiological Consequences

The ability of the emergency core cooling system to cope with the major loss-of-coolant accident (LOCA) is presented in Section 4.1 of this report. We have calculated that the consequences of the design basis accident assuming a TID-14844 fission product release and considering the effects of the spray system in reducing the iodine source in the containment. Results are presented in Table 7.6.

The design criteria for the control room are to limit the doses received by an operator continuously occupying the control room for 30 days following a loss-of-coolant accident to 5 rem whole body and 30 rem to the thyroid. However, the applicant has calculated the effectiveness of the shielding and ventilation system provided assuming (1) release of gap activity only, (2) iodine removal with a spray removal constant (λ) of 20.5 hr^{-1} for elemental iodine and 0.81 hr^{-1} for methyl iodine, (3) 5% methyl iodine, and (4) a wind speed of 2.5 m/sec. We will require that the control room meet the applicant's criteria with the following assumptions:

(1) TID-14844 fission product source, (2) spray removal constant of 2.5 hr^{-1} for elemental iodine, (3) 15% of the iodine is non-removable, and (4) a wind speed of 1 m/sec.

7.5.2 Hydrogen Evolution and Control

During the course of our review, the applicant has reduced the amount of aluminum used in his analysis of hydrogen evolution from 2000 lb to 500 lb. We have been informed orally that the total aluminum inventory will be held to 500 lb or less in recognition of the concerns with hydrogen and materials compatibility raised in connection with the Palisades POL review. We have informed the applicant that these matters are not considered resolved at this stage of the Midland review.

With regard to hydrogen buildup following a LOCA, the applicant calculates that a period of 36 days is available before it would be necessary to initiate his proposed containment purge system in order to control the hydrogen level at 3.5 v/o or below. Assuming continuous purging through 90% efficient iodine filters, the applicant estimates the incremental thyroid dose attributable to the purging operation at 1600 meters , the outer boundary of the low population zone, is 0.01 rem. The sources of hydrogen assumed include a 1% metal-water reaction, radiolysis, and corrosion of the 500 lb of aluminum.

Our assessment of the potential purge doses for the Midland site under somewhat more conservative assumptions than those of the applicant is shown in Table 7.5.2.

TABLE 7.5.2

TIME TO ATTAIN INDICATED H_2 LEVELS AND RESULTANT PURGE DOSE

		Hydrogen Sources[1]		Potential Purge Doses, Rem[2]			
				Exclusion Dist.		LPZ	
H_2 Level At Which Purge Is Initiated	Time to Reach H_2 Level (Days)	M/W Reaction Assumed (% Clad)	Aluminum Corrosion (lbs)	Thyroid	WB	Thyroid	WB
3.5 v/o	9	5%	500	407	24.4	54.4	3.2
4.0 v/o	16.4	5%	500	162	7.8	21	1
3.5 v/o	18.5	1%	500	148	6.1	19.4	0.8
4.0 v/o	29.3	1%	500	43	2.6	5.6	0.34

[1] Radiolysis assumptions as stated per 12/31/69 draft discussion paper to ACRS.

[2] Continuous purge, 4-30 day long term atmospheric diffusion, 2.5% of I-131 in non-removal gaseous form is available for purge, (consistent with use of internal cleanup from spray additives), 90% efficient iodine filtration in purge exhaust.

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As indicated in Table 7.5.2, the potential radiological consequences of hydrogen purging may be significant for the Midland Plant, particularly in view of the proximity of the city of Midland. The applicant however, has concluded that the "purge method is entirely suitable for preventing buildup of explosive hydrogen mixtures in the reactor building."

Consistent with our current approach to this matter, the applicant has been advised that purging may not be acceptable as the primary means of limiting the hydrogen buildup for the Midland Plant. He states that he is currently investigating certain other methods such as reactor building inerting, catalytic recombiners and flame recombiners to determine "feasibility of capacity, reliability and safety." He has not provided a definitive program including a completion schedule for this effort.

7.6 Radiological Consequences

As noted above, we have estimated the radiological doses for several accidents considered in the safety analyses. These are presented below in Table 7.6.

TABLE 7.6

ACCIDENT CONSEQUENCES

(Staff Calculations)

ACCIDENT	TWO HOUR SITE BOUNDARY DOSES 500 Meters		LPZ COURSE OF ACCIDENT DOSES One Mile	
	Thyroid	Whole Body	Thyroid	Whole Body
LOCA	270 Rem	4 Rem	280 Rem	4 Rem
REFUELING	250 Rem	8 Rem	90 Rem	3 Rem
ROD EJECTION	180 Rem	1 Rem	170 Rem	1 Rem
STEAM GENERATOR TUBE RUPTURE	250 Rem	2 Rem	90 Rem	< 1 Rem
GAS DECAY TANK RUPTURE	- -	12 Rem	- -	4 Rem

8.0 RESEARCH & DEVELOPMENT

Specific areas requiring research and development prior to design completion are summarized below.

8.1 Core Stability & Power Distribution Monitoring

The B&W program on xenon oscillations consists of the following analyses:

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

The results of the modal analysis have been submitted as Topical Report BAW-10010, Stability Margin for Xenon Oscillations-Modal Analysis. One dimensional digital analysis will be used to ascertain validity of the modal analysis approach. The results of the one and two dimensional digital analyses will be compiled as a topical report shortly. The results of the three dimensional digital analysis will be filed in the first quarter of 1970. The entire program is scheduled for completion well before the scheduled startup of Midland Plant Unit 1. This program is aimed at establishing the stability characteristics of the core and demonstrating, if necessary, that in the event of instability adequate control systems can be employed to assure the desired operation of the plant. This study will include the relationship between the indication of the out-of-core detectors and part-length rod position.

We are not sure if the B&W program will be able to demonstrate that sufficient information can be derived from external detectors alone. We have reservations that a known power distribution can be achieved after the reactor has been operated because axial burnup is not uniform. Further, fuel or control rod replacement or errors in fuel element position or enrichment may also perturb the flux distribution. In addition, we believe that insufficient experience exists with operation of large power reactors to ascertain when out-of-core detectors must be recalibrated. If the planned R&D program does not produce completely 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 available to the operator at all times when the reactor is operating at rated power.

8.2 Fuel Rod Clad Failure

B&W has initiated a study of fuel clad failure mechanisms associated with a loss-of-coolant accident which includes an evaluation of existing data and scoping tests to obtain data on potential failure mechanisms. These tests consist of the following:

1. Eutectic formation - test data indicate that a liquid eutectic forms at temperatures above 1700°F at the point of contact between the stainless steel spacer grid and the zircaloy clad. The applicant reports that nothing occurred which would interfere with emergency core cooling. The applicant has stated that work in this area is complete; we are expecting a report on this matter shortly.

2. Brittle failure - clad specimens heated to 2300°F at points in room temperature water did not experience brittle failure. A reduction in ductility occurred but strength was not reduced. The applicant has reported that work in this area is complete. We are awaiting a report on the experiments.
3. Clad swelling - single rod tests have been run to investigate the effects of clad swelling, the heatup rate, internal pressure, hydriding of Zircaloy, internal pressure, and preoxidation of the cladding. The applicant has reported that results to date indicate that: (1) the low pressure tests produced a larger increase in diameter due to greater ductility at higher temperatures, (2) the lower heat rates produce greater swelling, (3) the hydrogen content plays no major role in the event of diametral swelling, (4) the preoxidation generally resulted in less swelling. (On this basis, it was decided to delete the systematic study of preoxidation effects on swelling.) (5) the perforations were randomly located on the cladding, (6) the failure time is extremely short, (7) the first point of swelling was not necessarily the one which ruptured or swelled the greatest. Multirod experiments are planned using oven heating. A 4 x 4 bundle will be heated in an oven with the four central rods pressurized.

The analytical study of fuel clad failure is in the planning stage. This program will consist of evaluation of the axial as well as radial temperature distributions throughout the core. The change in flow channel resistance to flow was calculated and incorporated into the channel analysis. The 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. The multi-pin tests will provide data to determine the possible interaction between pins undergoing a temperature excursion. These data, coupled with the data resulting from completion of the FLECHT Program (Full Length Emergency Cooling Heat Transfer Test), scheduled for completion in 1970, will provide futher information on the capability of the emergency core cooling system to function as designed.

8.3 Internals Vent Valves

An experimental program has been performed by B&W on the internals vent valve assemblies, which included a hydrostatic test, valve disc closing test, a verification of pressure differences to open the valve discs and maintain the valve disc in the maximum open position, a functional handling test, vibration test, and test of prototype valves in a 1/6 scale model of the reactor vessel and internals. This test program has been completed and is reported in the proprietary B&W Topical Report BAW-10005. We are presently reviewing this report.

8.4 Once-Through Steam Generator

B&W has conducted tests on 7, 19, and 37 tube mockups of the once-through steam generator in the following areas: heat transfer and heat capacity, control and dynamic response, structural integrity under normal and accident conditions, vibration, feedwater heating by spray nozzles, tube leakage propagation, and simulated steamline failure tests. This program is complete and is reported in BAW-10002. We are reviewing this report at the present time and have identified areas where further justification must be supplied before we can accept the B&W conclusion that the tests substantiate the design. Our discussions with B&W are being conducted in the course of our review of the Oconee FSAR.

8.5 Reagent Spray System

B&W has performed tests on sodium thiosulfate stability while stored and under accident conditions. Further, material compatability 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. Tests are planned on welded samples and should be completed during the third quarter of 1970. As noted in Section 4.3, materials compatability testing, including welded specimens, must be completed before we can assess the long-term stability of the sodium thiosulfate solution in the post-loss-of-coolant accident environment. We have informed the applicant that space should be reserved for charcoal filters should the R&D program fail to meet its objectives.

8.6 Process Steam Monitoring

In order to determine the adequacy and feasibility of the process steam monitoring system, the applicant will conduct a research and development program consisting of the following:

1. Tests to determine decontamination factors on various pieces of equipment which will come in contact with the process steam. These tests will be conducted at the Big Rock Point plant.
2. Verification that the proposed monitor can detect N-13 activity under simulated operating conditions at the Palisades plant.

These tests will be completed prior to the submittal of the FSAR.

9.0 TECHNICAL QUALIFICATIONS

We have reviewed the application with respect to the adequacy of the technical qualifications of Consumers Power Company and its contractors to construct the facility. The execution of the project is the sole responsibility of Consumers Power. They have previous nuclear experience through their operation and construction of the Big Rock Point Plant and construction of the Palisades Plant.

Consumers Power has engaged the Babcock & Wilcox Company to design and supply both 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 will perform the onsite construction of the plant. On the basis of our previous and current evaluations of plants designed and constructed by the contractors, and the applicant's experience in operation of Big Rock and construction of both Big Rock and Palisades, we conclude that Consumers Power Company and its contractors are technically qualified to design and construct the Midland Plant.

10.0 STAFFING

Consumers Power Company will review the plant design, equipment selection, and construction and will participate in acceptance testing as construction progresses. The applicant is currently operating the Big Rock Point Nuclear Station and completing the final phase of construction of the Palisades Nuclear Plant. During construction of the facility, we will monitor Consumers' capabilities through the Division of Compliance to ensure that this expanding commitment to nuclear power does not dilute the technical support organization.

The onsite plant organization closely parallels that of the Palisades organization, with three main groups under the general direction of the Plant Superintendent. These are the Maintenance, Technical Support, and the Operations Groups. The general plant organizational arrangement is satisfactory. However, the proposed dual-unit shift composition of one Senior Licensed Operator, three Licensed Control Operators and three Auxiliary Operators per shift is inadequate, unless the applicant can demonstrate his ability to safely handle both normal and abnormal conditions at the facility. We have expressed orally our concerns in this area to the applicant to permit him to adjust his training program in order to assure adequate staff capability at the operating license stage, when this point will have greater significance. The applicant's minimum qualifications for plant personnel will be in accordance with Section 4 of the Proposed Standard for Selection and Training of Personnel for

Nuclear Power Plants prepared by the ANS-3 Committee, Draft No. 9, or any subsequent approved revision. We consider this satisfactory.

The plans for training the staff to meet the above qualifications are based primarily on the fact that the supervisory personnel will receive their major training at either Big Rock Point or the Palisades Plant and that a significant number of the control operators will come from the Big Rock Point or the Palisades Plant Organizations and will hold operator licenses on the plant at which they were previously assigned. We consider these plans to be marginally acceptable, and will require considerably more detail concerning the training program at the OL review. We have expressed our concerns in this area to the applicant.

11.0 EMERGENCY PLANNING

In the event of a radiological emergency, the Consumers shift supervisor at the nuclear facility will undertake the following actions in the order listed:

1. Make certain the reactor is shut down and in a safe condition.
2. Make certain that containment isolation valves and ventilation valves are closed.
3. Utilize monitoring equipment to determine radiation levels within the plant and at the plant boundary fence.
4. Check for possible missing or injured personnel.
5. Dispatch personnel to the access road entrance to control access.
6. Establish a personnel monitoring and change station along the reactor plant evacuation route.
7. Evacuate Consumers personnel as necessary.
8. Continue manning control room.
9. Notify Dow Chemical Company plant protection supervisor.
10. Notify the plant superintendent or the assistant plant superintendent if the plant superintendent is unavailable.
11. Notify the Radiation Protection Supervisor.
12. Notify the other plant personnel as necessary.
13. Notify selected Consumers Power Company management personnel.

14. Notify civil authorities. (This will be initiated by either the plant superintendent or the assistant plant superintendent. If they are unavailable, the shift supervisor will initiate notification provided that (1) radiation levels are in excess of 1r/hr at the plant boundary fence and (2) danger might exist to persons or property outside the Consumers property boundary.)

Although only a small portion of the Dow plant now lies within the exclusion area, Dow has agreed to evacuate their site in the event of a radiological emergency, if ordered to do so by Consumers Power Company. Our concern regarding the prompt evacuation of personnel from the Dow property, as expressed in our January 23, 1969, report to the Committee has been ameliorated by the installation of reagent sprays for iodine removal and a reduction in containment leakage rate. (The potential 30-day thyroid dose following a LOCA at 1600 meters has been reduced from 3,000 rem to 280 rem by these modifications.)

The Dow emergency plan is identical to that described in our earlier report. Upon receipt of the call from Consumers Power Company (Step 9), the Dow Chemical Company Telephone Alert System would be activated. The following actions would then take place:

1. The Dow dispatcher writes the message on a Telephone Alert System form.

2. The dispatcher actuates the alert system by depressing a single key and by telephone simultaneously notifies the 700 units within the plant of the emergency.
3. The telephone message is repeated for a minimum of two minutes to permit time to answer phones and receive the message.

As the individual units received the message, each would sound its siren continuously, institute a crash shutdown of operations, and evacuate the site. Time intervals of up to 45 minutes would be required to terminate the chemical processes and under these circumstances some of the Dow people would remain on duty until the processes are shut down. The Dow estimate of the time required to evacuate the site by the personnel involved in the various processes indicates that 90% of the Dow personnel can be evacuated within 20 minutes, and all can be evacuated within 45 minutes of receipt of the evacuation signal at the process units. Most employees are located from one to three miles from the facility. Use would be made of available department vehicles in transporting personnel to the parking lots located approximately 1/2 mile from the center of the site.

We have estimated the potential doses which might be received following a LOCA at one mile from the reactors during a 35-minute period to be 55 rem. Considering delays in initiating evacuation and the fact that most Dow employees are located from one to three miles from the reactors, this represents an estimate of the maximum dose which might be received by the 90% of the Dow personnel who evacuate within 20 minutes. We have also calculated the 1-hour dose at one mile to be 75 rem. We consider this to be representative of the maximum potential dose which might be received by those Dow employees who must remain onsite to shut down Dow facilities.

Based on the above, we conclude the Dow evacuation plans are adequate to protect Dow employees.

11.3 Plant Security

The cooling pond dikes and the plant site will be enclosed by a security fence. The top of the dike will be paved and the fenced-in area will be patrolled on a once per shift basis. The immediate plant area will be checked once per hour by an operator in the course of checking outdoor equipment. All fence gates will be normally locked and will be attended when open. The main gate is operable from the control room and identification will be required before entrance is permitted. All outside doors will be normally locked after dayworkers have left the site. Since access to the facility is restricted and important areas will be checked at least once per hour, we conclude the plant security measures are acceptable.

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Docket Nos. 50-329 &
50-330

September 11, 1970

Report No. 5 to the ACRS

MIDLAND PLANT UNIT NOS. 1 & 2
CONSTRUCTION PERMIT REVIEW

U. S. Atomic Energy Commission
Division of Reactor Licensing

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