CONSUMERS POWER COMPANY

APPLICATION FOR

REACTOR CONSTRUCTION PERMIT AND OPERATING LICENSE

Docket No. 50-329
Docket No. 50-330

Regulatory File Cy.

Amendment No. 12

Received w/ltr Dated 5-28-70

Enclosed herewith, amending and supplementing the above-entitled application, are the following:

1. Revised and additional pages in response to questions raised by the ACRS at the meeting held May 8, 1970. These pages deal with reactor vessel integrity, analysis of hazardous chlorine release and ground surface subsidence. Revised and additional pages bear the notation "Amendment No. 12, 5/28/70," except as noted below and are marked in the margin to indicate where changes have been made. The enclosed pages replace existing pages submitted in Amendment No. 11 (dated May 1, 1970) having the same numbers except that pages marked with an asterisk in the list below are in addition to and do not replace existing pages.

PAGES

12.00-1*	2.17-10a*
2.00-3	Dow Letter* (No Notation)

2. Revised pages in response to conversations with the DRL staff by telephone and at meetings May 25 and May 26, 1970. These pages deal with high-pressure level alarm integrity and radioactivity in the process steam. Revised pages bear the notation "Amendment No. 12, 5/28/70," and are marked in the margin to indicate where changes have been made. The enclosed pages replace existing pages submitted in Amendment No. 11 (dated May 1, 1970) having the same numbers.

PAGES

7.00-1	11.00-2
11.00-1	11.00-3

CONSUMERS POWER COMPANY

By R. C. Youngdahl
R. C. Youngdahl, Senior Vice President

Dated May 28, 1970

Sworn to and subscribed to before me this 28th day of May, 1970

Helen R. Lehr
Notary Public, Jackson County, Michigan
My Commission Expires December 11, 1973

12. Reactor Vessel Integrity

Consumers Power Company has addressed itself to the possibility of breaching the reactor vessel while at pressure. This possibility was discussed by CPCo with the major United States vessel manufacturers, a consultant, and the A-E, Bechtel. Based on a technical assessment of this possibility, including fracture mechanics, code requirements, Q-A procedures, in-service inspection, surveillance specimen program and a capability to reanneal the vessel, it is CPCo's conclusion that for the most adverse conditions which could be encountered throughout the Plant operating life, a breaching of the reactor vessel under a pressurized condition in a manner which would jeopardize public safety is not credible.

However, based on conclusions reach by the ACRS on February 8, 1969 and DRL as identified in P. A. Morris' letter to CPCo dated March 28, 1969 relating Midland to the Zion and Indian Point-3 Plants, the design of the reactor cavity will afford an additional measure of conservatism in that it will be capable of withstanding the mechanical forces and pressure transients comparable to those considered in the design of the Zion and Indian Point-3 Plants.

(4) The maximum calculated chlorine concentration of 3.6 ppm will be obtained in about 9 minutes after the cloud arrives (49 minutes after the release from the tank). This concentration is due to infiltration into the room.

(5) In about 90 minutes after the chlorine arrives, the concentration is to be reduced to less than 1 ppm. At this time, the Scott Airpacks can be removed and the filter system changed from the recirculation mode to a once-through mode to bring fresh outside air into the room.

(6) The chlorine loading on the charcoal will be about 8 percent of the weight of the charcoal. The maximum recommended loading is 15 percent.

(7) In the event chlorine leakage from the pit should continue at a rate of 78 lb/minute until all is evaporated (~30 days), the charcoal loading would only be about 10 percent of the weight of the charcoal.

Conclusions

The preceding study has shown that a major rupture of the chlorine tank would not require operating personnel to evacuate the Plant control room. Thus, safe operation of the reactor is assured.

Consumers Power Company will continue to study this potential problem and has included in its criteria for design of the control room, the objective of finding a practical method for maintaining the control room atmospheric chlorine concentration below the 8-hr TLV of 1 ppm even for the most serious conceivable chlorine accidents. The Scott Airpacks will be used as backup if such a method for chlorine removal is determined to be feasible and hence is incorporated in the Plant.

The applicant has determined that subsidence which could cause surface rupture will not occur at the Midland plant site due to salt mining or brine removal operations. However, to provide additional conservatism to the plant design criteria, the applicant and the Dow Chemical Company further agree to exclude all future salt mining operations from areas within an 0.5 mile distance from the Midland plant. (See attached letter.)

Extraction of brine and the reinjection of the waste brine into the Sylvania Sandstone occurs over a large area, and therefore exclusion of the brine operations is not included.

A review of salt well locations indicate that Salt Wells #9, 10, 17, 19 and 20 are included within this 0.5 mile radius. As noted in the General Analytics report, Well #10 is abandoned and plugged, Well #9 is shut down and no longer used, and Well #17 will be abandoned and plugged. Wells #19 and 20 are presently used as standby wells, but it is agreed to abandon and plug these wells to provide the indicated exclusion area.

THE DOW CHEMICAL COMPANY
MIDLAND DIVISION
MIDLAND, MICHIGAN 48640

May 22, 1970

Mr. W. E. Kessler
Consumers Power Company
212 W. Michigan Avenue
Jackson, Michigan 49201

Dear Mr. Kessler:

The Dow Chemical Company will not drill any brine wells or salt wells within one-half mile of the Consumers Power Midland Nuclear Plant after this date.

The status and conditions to be maintained for existing salt wells within one-half mile of the nuclear plant are:

No. 10 salt well is plugged and abandoned.

No. 17 salt well will be plugged and abandoned within 60 days.

No's. 9, 19, and 20 salt wells are in stand-by status, used only on occasion of emergency outage of active salt wells. They will be abandoned and plugged before 1975.

Sincerely yours,

James F. Maddox
Nuclear Energy Project
Basic Operations
256 Building

cc: R. E. Reinker
R. C. Hultin

elh

7. High-Pressure Level Alarm Integrity

While it has been shown that a pressurizer level trip is not required because the reactor is safe without such trip, the pressurizer water level alarm system provided will be such that failure of a single component in that system will not prevent proper alarm when required. Single component failure includes such events as the shorting or open circuiting of interconnecting signal or power cables. The criteria for the design of the pressurizer level alarm are as follows:

1. The two channels shall be independent and physically separated to reduce the possibility of interactions between channels during maintenance or channel malfunction.

2. The alarm channels shall be designed to maintain necessary functional capability under extremes of condition relating to the environment, energy supply, malfunctions and accidents.

3. Capability shall be provided for testing and calibrating channels during power operation.

4. The two alarm channels shall be tested on a periodic schedule including a sensor check. The sensor check shall consist of comparing redundant level measurements against each other.

5. In addition to its prime function, an alarm channel shall initiate its alarm function whenever a channel is placed on test or removed from service for maintenance.

In the event one of the two alarm channels fails, Plant operation will be permitted to continue provided operability of the remaining channel or availability of level information to the operator is verified by performing the normal on-line tests.

6. The above criteria apply up to and including the contacts which actuate the two annunciator boxes.

11. Radioactivity in the Process Steam

The Midland Nuclear Power Plant is designed to supply process steam to the Dow Chemical Company. In the event of a primary to secondary leak in the steam generator supplying this steam, radionuclides could be transported to Dow where physical barriers prevent the steam or its condensate from contacting or entering into any products.

The concentration of radionuclides in the process steam to Dow Chemical Company will be monitored by a continuous on-line gross gamma monitor (described in Item 12, Amendments 6 and 8) and by radiochemistry analysis of grab samples of the process steam condensate. These samples will be used to provide an indication of steam generator leakage, to provide a control on the yearly average activity in the process steam, and to provide an indication of when the process steam must be terminated or taken from another steam generator. The activity levels for three conditions are as follows:

Condition 1. The continuous monitor will provide an alarm when the gross gamma activity in the steam reaches 3×10^{-6} μ ci/cc. This is a warning and is not a level at which the process steam from a particular steam generator will be terminated.

Condition 2. The yearly average gross beta activity as collected from grab samples will not exceed 1/10 of MPC, as given in 10 CFR 20, for any radionuclide excluding noble gases and those with half-lives less than 12 hours.

Condition 3. For a limited period of time, the radioactivity in the steam may approach the MPC as determined by the equation:

$$\frac{C_A}{MPC_A} + \frac{C_B}{MPC_B} + \ldots . \leq 1, \text{ however,}$$

steam flow will be either terminated or the source of steam will be switched to an alternate steam generator when this limit is reached.

The radionuclide concentration in the process steam and the predicted yearly releases to the environment in the liquid discharged from the process steam condensate for each of the above three conditions are as follows:

A. Operation With 1% Failed Fuel

Condition 1. When a gross gamma monitor reading of 3×10^{-6} μci/cc is obtained, all radionuclides are at least a factor of 10 below their respective MPC, except for I-131. The iodine-131 concentration is predicted to be approximately a factor of 7 to 10 below its MPC. The gross beta activity, excluding noble gases, tritium and radionuclides with half-lives less than 12 hours, as determined from the grab samples, is predicted to be approximately 2.8×10^{-7} μci/cc. If the plant was operated for a year with this activity level, the total activity release from Dow to the river in the liquid from the process steam is predicted to be approximately 1.6 curies based on Dow's predicted release to the river of 1.5×10^{6} lb/hr condensate. Based on the predicted reactor coolant concentrations shown in Section 11 of the PSAR, it is predicted that a leakage rate of about 2×10^{-4} gpm will result in a gross gamma level of 3×10^{-6} μci/cc in the steam.

Condition 2. If the radionuclide concentration in the reactor coolant is as shown in Section 11, this case will be equivalent to Condition 1. That is, the yearly average gross beta activity will be approximately 2.8×10^{-7} μci/cc and the yearly release will be about 1.6 curies. However, in the event the iodine-131 concentration is lower than predicted, a larger average steam generator leakage rate may be permitted before the controlling radionuclide is 1/10 of its MPC. The yearly release, however, will remain at about 1.6 curies.

Condition 3. For this condition, it is predicted that the gross gamma activity as determined by the continuous gross gamma monitor or the gross beta activity as determined by the analysis of grab samples will be increased by approximated factor of 7. The iodine-131 will approach its MPC value, however, all others are predicted to be at least a factor of 10 below their respective MPC's.

B. Operation With No Failed Fuel

Condition 1. A gross gamma level of 3×10^{-6} μci/cc measured by continuous monitor is predicted to be equivalent to a grab sample gross beta activity of approximately 1.6×10^{-7} μci/cc excluding tritium, sodium ($T_{1/2}$ = 15 hrs) and radionuclides with half-lives less than 12 hours. The total yearly release based on the 1.6×10^{-7} μci/cc is predicted to be approximately 0.93 ci based on the assumed Dow release to the river of 1.5×10^{6} lb/hr condensate. The

tritium concentration is predicted to be a factor of about 30 below its MPC and all other radionuclides with half-lives greater than 12 hours will be at a factor of 100 below their respective MPC's.

The predicted generator leakage rate required for these activity levels is approximately 0.2 gpm.

Condition 2. The yearly average gross beta activity in the steam will not exceed 2.2×10^{-7} μci/cc excluding tritium, sodium and other radionuclides with half-lives less than 12 hours. For this concentration the predicted annual release will be approximately 1.6 ci, again based on the assumed release of 1.5×10^{6} lb/hr of condensate. This annual release is the same as for the 1% failed fuel case. It is also predicted that the tritium concentration will be at least a factor of 20 below its MPC and all others with half-lives greater than 12 hours will be at least a factor of 100 below their respective MPC's.

Based on the radionuclide concentration of the reactor coolant (Section 11 of the PSAR), it is predicted that a steam generator leakage rate of 0.28 gpm would provide this activity level in the steam. However, in the event the specific radionuclide concentrations are less than those shown, the allowable leakage rate could be increased to provide a yearly average of 2.2×10^{-7} μci/cc or until a maximum leakage rate of 1 gpm is obtained. Also, if it is found that the specific radionuclide concentrations are larger than those shown in PSAR Section 11, the maximum allowable leakage rate would be decreased in order not to exceed the yearly average concentration of 2.2×10^{-7} μci/cc. The total yearly release would remain at 1.6 ci.

Condition 3. For short-term operation, the radionuclide concentration in the process steam will not exceed the concentration in Condition 2 above by more than a factor of approximately 3.6. This is based on a maximum steam generator leakage of 1 gpm.

The tritium concentration is predicted to be at least a factor of 4 below its MPC and all other radionuclides with half-lives greater than 12 hours will be at least a factor of 10 below their respective MPC's.

The annual release will remain the same as for Condition 2.

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

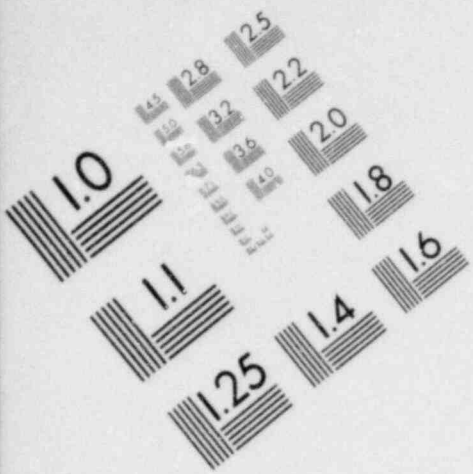
In the Matter of	)	
Consumers Power Company	)	
Application for Reactor	)	Docket No. 50-329
Construction Permit and	)	Docket No. 50-330
Operating License	)	

CERTIFICATE OF SERVICE

Amendment No. 12 to Consumers Power Company's Application for Reactor Construction Permit and Operating License for Midland Plant Units Nos. 1 and 2 has been served today on Robert B. Chatterton, Township Supervisor, Midland Township, Midland County, Michigan, by deposit in the United States mail.

J. E. Rice
Attorney
Consumers Power Company

Dated: May 28, 1970



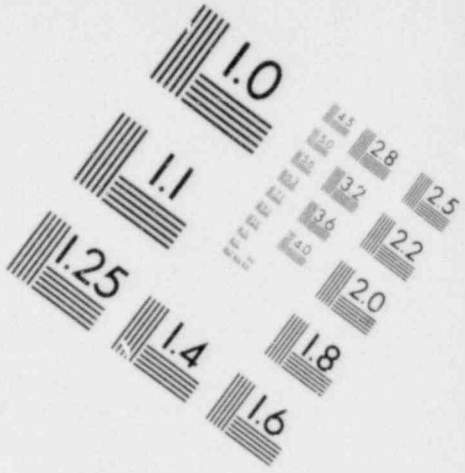
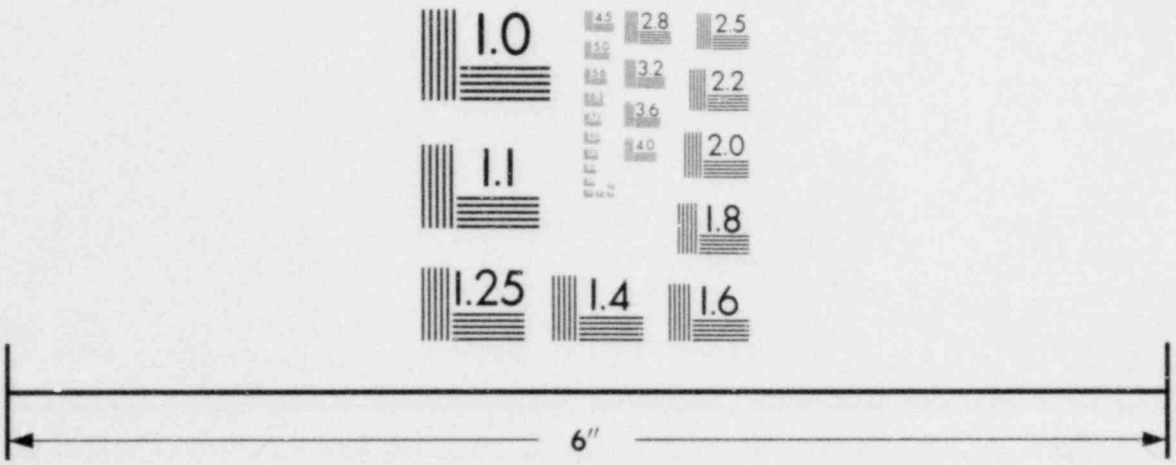
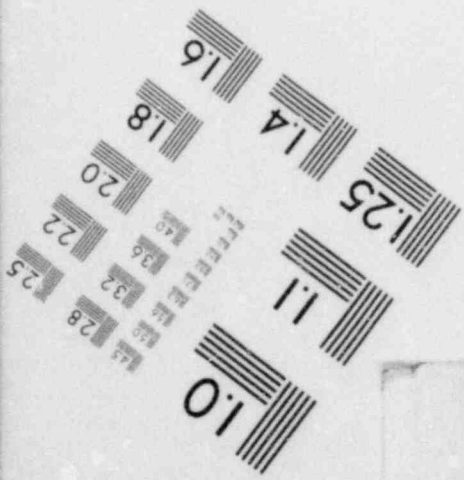
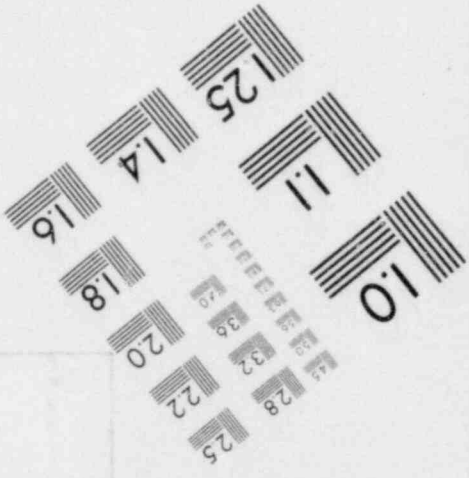


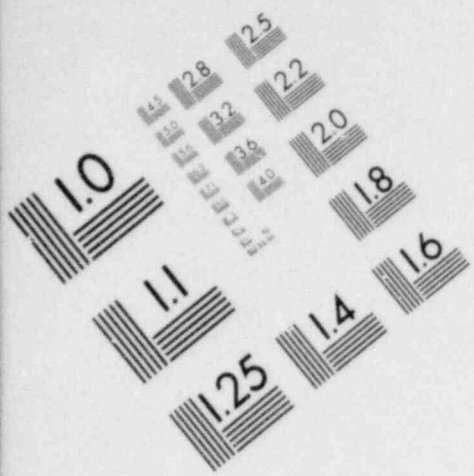
IMAGE EVALUATION
TEST TARGET (MT-3)



MICROCOPY RESOLUTION TEST CHART







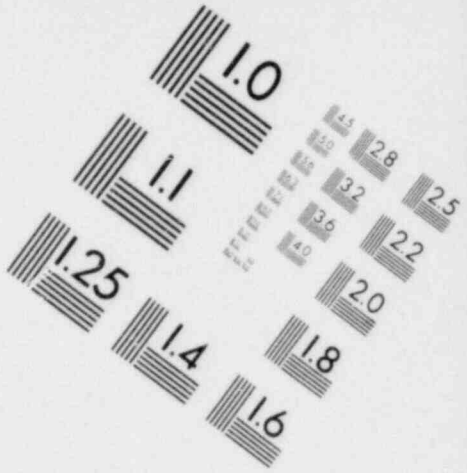
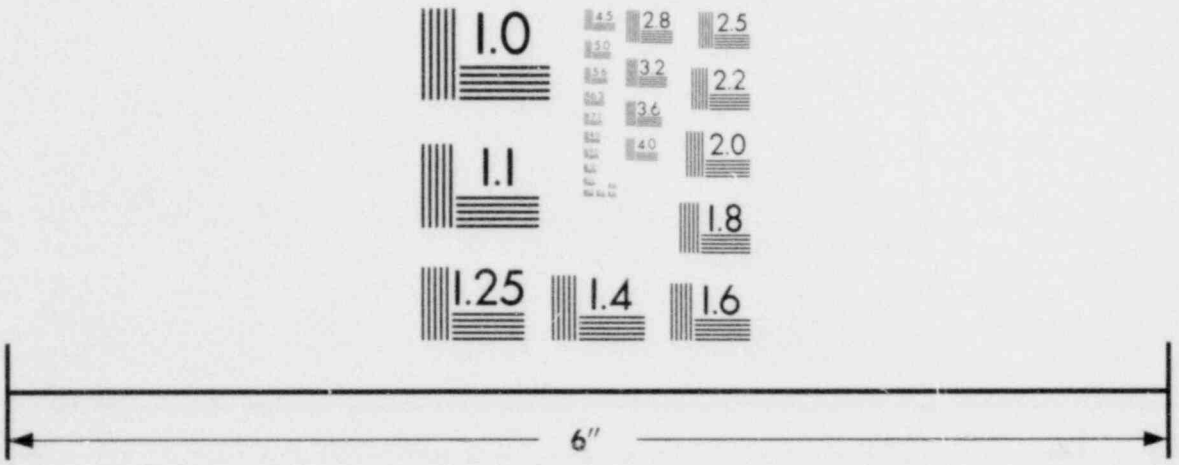


IMAGE EVALUATION TEST TARGET (MT-3)



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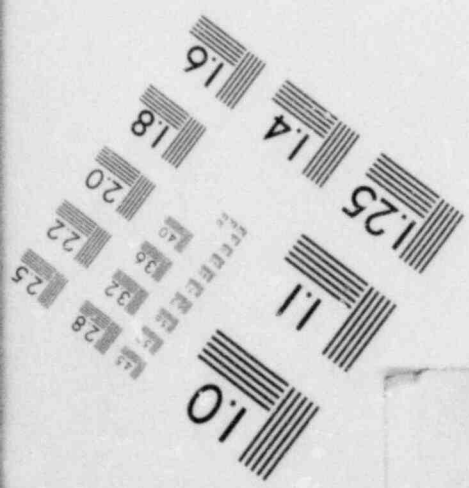

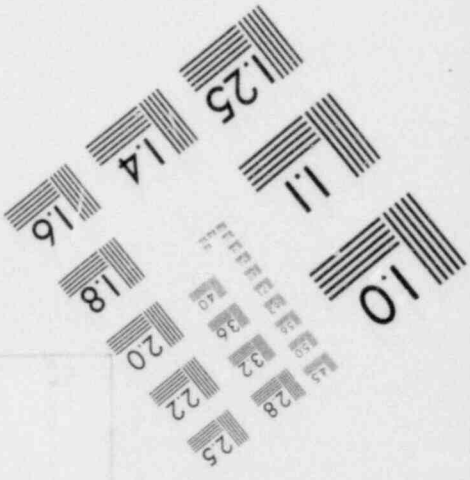

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

In the Matter of	)	
Consumers Power Company	)	
Application for Reactor	)	Docket No. 50-329
Construction Permit and	)	Docket No. 50-330
Operating License	)	

CERTIFICATE OF SERVICE

Amendment No. 12 to Consumers Power Company's Application for Reactor Construction Permit and Operating License for Midland Plant Units Nos. 1 and 2 has been served today on Frank Olds, Chairman of the County Board of Supervisors, Midland County, Michigan, by deposit in the United States mail.

J. E. Rice
Attorney
Consumers Power Company

Dated: May 28, 1970

FROM: Consumers Power Co.
Jackson, Mich.
R.C. Youngdahl

DATE OF DOCUMENT: 5-28-70
DATE RECEIVED: 6-1-70
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CONCURRENCE ☐ | COMMENT ☐
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3 sigend & 87 conf'd cys

ENCLOSURES:
Amdt # 12, notarized 5-28-70, submitting addtl info..re: ACRS meeting 5-8-70, & DRL meeting 5-25-26-70....consisting of revised pages for PSARept

CERTIFICATE OF SV, dtd 5-28-70, showing sv of Amdt # 12, upon Robert B. Chatterton & Frank Olds.

REMARKS:

(18) cys holding for ACRS

REFERRED TO	DATE	RECEIVED BY	DATE
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