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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket No. 50-155

OCT 2 7 1964

Consumers Power Company
1945 Parnell Road
Jackson, Michigan

Attention: Mr. H. R. Wall
Vice President

Gentlemen:

The Division of Reactor Licensing has reviewed the information contained in your TWX dated September 28, 1964 concerning the main steam bypass valve failure and resultant primary system blowdown which occurred on September 18 during stability testing to evaluate recent unexplained power oscillations during the high power density test program. Copies of the TWX have been transmitted to the Advisory Committee on Reactor Safeguards for its information.

As a result of our review we have concluded that additional information is required to adequately analyze the safety significance of these occurrences and the subsequent control rod drive system malfunctions. The ACRS, at its October meeting, also indicated an interest in further evaluation of the occurrences described in the TWX. Accordingly, you are requested, pursuant to Paragraph 50.54(f) of the Commission's regulations, to provide the information indicated in the attached list within 30 days of receipt of this letter. Forty copies of the information requested should be provided.

We plan to schedule a meeting shortly after receipt of the information requested in order to discuss these occurrences with your representatives and members of the ACRS Subcommittee on Big Rock

REGISTERED MAIL
RETURN RECEIPT REQUESTED

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Point. We will advise you later of the suggested date and place of these discussions. In the meantime, should you have any questions concerning the information requested, we would be pleased to meet with you at your convenience.

Sincerely yours,

Original Signed By
R.L. Doan

R. L. Doan, Director
Division of Reactor Licensing

Attachment:
Additional Information Requested

cc: Dr. Herbert Kouts, ACRS

CONSUMERS POWER COMPANY

ADDITIONAL INFORMATION REQUESTED

1. Core Oscillations

1.1 A comparison between the calculated reactor stability, in terms of phase margin in degrees and gain margin in decibels, and the experimentally determined values. The sensitivity to changes in flow recirculation, pressure, and power should be shown for the 60 KW/liter average power density case, and other test conditions where the experimental results are readily available. The dependence of stability on control rod position or power distribution should be compared for two conditions, one where all control rods are fully withdrawn, except C3 which is withdrawn to about 18 inches, and a second condition where all control rods are withdrawn, except C3 which is fully inserted, and B2 and D4 which are withdrawn to 18 inches.

1.2 An analysis and evaluation of the reactor core oscillations, initially observed on August 24, 1964, after a long period of stable operation, supplemented by appropriate test results of additional high average power density runs performed on the reconstituted 44 bundle core. Conclusions based on experimental data obtained from the Big Rock plant should be stated along with supporting information, particularly noting the apparent causes for the observed oscillations. To insure a better understanding of data presented, important parameters such as temperature, rod position, pressure, power level, recirculation flow, feedwater temperature, etc., should be identified.

2. Main Steam Bypass Failure

2.1 A graphical presentation of the following plant data as a function of time over the 103 second transient:

a. Three picoammeter signals
b. Two intermediate range signals
c. Five in-core ion chambers (identify each)
d. Pressure
e. Core inlet temperature
f. Core exit temperature
g. Recirculation flow rate
h. Feedwater temperature
i. Feedwater flow rate
j. Electric power output
k. Steam drum level

2.2 The calculated or absolute power level experienced during the transient. The method and basis therefor used to determine "true" reactor power during the transient or more specifically at the time of maximum indicated power should be provided.

2.3 The relative importance of moderator and reflector density changes on the power range detector signal attenuation. Quantitative analyses should be provided to demonstrate the accuracy or lack thereof of installed external power level instrumentation in measuring reactor power level. The correlations or experimental data used to determine attenuation factor moderator-reflector density relationship should be provided.

2.4 Reactor power level determination by in-pile detectors. In view of the changing power distribution within the core as more voids are formed due to reduced pressure, the significance of the local power detector signals, as indicators of total reactor power, should be explained.

2.5 Recirculation coolant transport time. It has been reported that the feedwater was being controlled manually at the time of the accidental valve opening. The reduced reactor coolant pressure and preheat caused an increase in feedwater flow and reduction in temperature. This effect on core reactivity should be analyzed along with the reduced core temperature resulting from the reduced pressure e.g., although more voids were formed in the upper reactor region the lower core region temperature decreased because of reduced saturation temperature and an increase in colder feedwater.

2.6 A description of the analytical methods, assumptions, and correlations used in the transient thermal analyses. Earlier calculations of fuel centerline temperatures at various transient power levels were made on a steady state basis. From recent correspondence, based on 100% steady state power level where calculated heat flux is 381,000 BTU/hr $ft^2$ and center fuel temperature is 3170°F, it appears that transient analytical methods are now being used.

2.7 A discussion of the basis for your conclusion that the incident has caused no structural changes in the fuel, the reactor vessel, vessel internals, and primary system.

2.8 Circumstances and causes leading to the accidental opening of the main steam bypass valve and failure of the remote manual override switch. We understand that a similar failure of the main steam bypass valve has occurred previously which resulted in the installation of the remote manual override switch so that the bypass valve could be reclosed before excessive steam blow down. A description of the changes or improvements being implemented to prevent a recurrence of such an incident should be provided.

3. Malfunctioning Control Rods

3.1 Revised operational limits and procedures which we understand have been placed in effect in order to prevent excessive galling when exercising of the control rods shows evidence of high friction. In the event of recurrence of a sticking control rod, will alternate rod programs be developed to permit a new rod withdrawal sequence until the rod or rods with high friction can be investigated and corrected or, will it be necessary to shutdown the reactor? How will "excessive" control rod friction be defined?

3.2 A description of the new control rod blades which we understand are to be installed at Big Rock. Will the new blades have Stellite rollers? Does the absence of one or more Stellite rollers in the present design allow the control blade to wobble or vibrate excessively?

3.3 An analysis of crud developed during reactor operations to date. Has it been determined experimentally that crud is hard enough to gall the index tube?

4. Thermal Shield Bolt Failures

4.1 A discussion of the thermal shield bolt failures, including the results to date of the investigation of the causes of the occurrence.

4.2 An evaluation of the relationship between this occurrence and the other problems recently encountered.

4.3 A description of any proposed modifications to prevent recurrence of these failures.