Regulatory File Cy.

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Commonwealth Edison Company

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July 31, 1970

Dr. Peter A. Morris, Director
Division of Reactor Licensing
U.S. Atomic Energy Commission
Washington, D.C. 20545

Subject: Additional Information Relative to Provisional Operating License DPR-19 for Dresden Unit 2

Dear Dr. Morris:

The purpose of this letter is to provide you with information concerning recent operational difficulties on Dresden Unit 2 and our proposed corrective actions. This information has been requested in your letter dated June 18, 1970 which responded to our proposed Change No. 2 to Appendix A of DPR-19.

Attached hereto is Exhibit I containing our evaluation of the following operational difficulties:

1. Progress to date on investigative program on unexplained fuel failures;

2. Results of investigative program on excessive control rod scram times.

The other information requested in your letter of June 18, 1970 was submitted by our letter dated July 6, 1970 and Exhibit I attached thereto.

In addition to three signed originals, 19 copies of this information are also submitted.

Very truly yours,

Byron Lee Jr.

Byron Lee, Jr.
Assistant to the President

SUBSCRIBED and SWORN to before me this 31st day of July, 1970.

Patricia A. Nelson
Notary Public

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DRESDEN II

INVESTIGATION OF FUEL DEFECTS

July, 1970

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July 29, 1970

Dresden II had essentially completed the scheduled startup testing at 75 percent of rated thermal power, and preparations were underway to achieve full rated power when, on June 5, 1970, an unplanned transient occurred resulting in steam safety valve discharge into the drywell. The events leading to and following this transient have been described elsewhere and will not be discussed in this report. The extended shutdown for drywell cleanup and resetting of safety valves was utilized to remove the reactor vessel head and investigate the extent of prior fuel failures.

Significant offgas release had been observed as early as the first week of May during operation and testing at 50 percent of rated power. This offgas release persisted and increased briefly to a maximum of 100,000 microcuries per second at 75 percent power by the end of May. Insertion of eight control rods on two groups of four adjacent rods reduced the offgas release rate to approximately 40,000 microcuries per second and, during one brief period of operation, the release rate was as low as 17,000 microcuries per second at 75 percent power.

Prior to the extended shutdown, further operation was contemplated with these eight rods inserted in order to minimize the offgas release rates. The eight rods were selected as the result of "flux tilting" measurements. In these flux tilting measurements, control rods were inserted from the normal pattern individually or in small groups, and the resulting change in offgas release rate was noted. Flux tilting indicated a total of 16 control rod cells, or 64 fuel assemblies, as the most probable suspect areas for the location of failed fuel assemblies.

It should be noted that the flux, and therefore the offgas release rate, at a failed fuel assembly may be affected to a varying degree by the repositioning of many control rods. Considerable interpretive judgment of the results is required and the results are somewhat imprecise. In addition to the overlapping effects of repositioning any of several control rods near a failed bundle, there is the inability to distinguish

which of four bundles in a cell is the failed assembly. Therefore, the next and most conclusive step in failed fuel location is "sipping" or sampling of stagnant cooling water in the vicinity of the fuel for the presence of fission products.

Sipping was initiated as soon as possible after the head and core internals were removed. However, the total elapsed time from the previous operating period was greater than three weeks and in-core sipping effectiveness was considerably reduced. Ordinarily, in-core sipping is performed within two to three days of shutdown, or as soon as possible, in order that the short-lived fission products may be detected before significant decay and also to catch the core with considerable decay heat to aid the sipping process by driving off the fission products. Another factor which made in-core sipping less effective was that plant rated power has not been achieved and operation at 75 percent power was for short periods of time. Under these conditions the longer lived fission products which would have provided most of the signal and the decay heat after the longer shutdown time were not permitted to build up toward equilibrium values. In particular, cesium was not present in the abundance that would be observed in an equilibrium core.

In-core sipping is performed by placing a fixture resembling an inverted funnel over a cell of four fuel assemblies and pressurizing this with air. This stops the cooling water flow to these four assemblies and the decay heat drives fission products out through any existing cladding defects. Water samples are pulled from the stagnant volume enclosed by each of the four assemblies. The radioactivity in each sample is compared with the general background in the reactor water and significant increases over background are interpreted as indications of defective cladding. Out-of-core sipping is performed by loading the fuel assemblies one at a time in an enclosed can, flushing with clean water to reduce the general background level and permitting the assembly to soak for several hours.

This technique is more sensitive than in-core sipping because the background can be reduced and all fission products released during the soaking period can be collected instead of dispersing through the core. However, it is considerably more time consuming because of the individual handling requirements for each fuel assembly and because of the longer soak time and, under more normal conditions, is used primarily for final verification of in-core sipping results.

During the recent Dresden II outage, a total of 131 fuel assemblies were sipped out of core. In addition to the assemblies selected on the basis of flux tilting data, assemblies that were removed for other reasons were sipped while in storage in the fuel pool. In this manner a somewhat random sampling of the core was obtained. A total of 27 fuel assemblies were identified as failed on the basis of sip signals. Two others remained out of core because of visual inspection results, for a total of 29 fuel assemblies. These 29 assemblies were replaced with identical new assemblies that had been fabricated for Dresden III.

Sipping was supplemented by visual inspection, using underwater TV and a borescope. Channels were removed from the assemblies in the fuel preparation machine, and each side of the assembly was examined full length by raising and lowering the assembly past the fixed TV camera and borescope. Less than half of the sipping leakers exhibited visual defects during this inspection.

Special tools were fabricated to permit underwater disassembly of the fuel bundles and examination of individual rods. Four bundles were completely disassembled and individual rod failures examined. The defects that were observed were minor, being primarily small blisters on individual rods. These blisters indicate a highly localized chemical

reaction in the cladding. The cause has not been determined at this time, but it is most likely the result of an abnormal condition introduced in the manufacture of the fuel. The localized points of reaction are brittle. Perforation of the clad at the blister is probably accelerated by power and thermal cycling of the fuel, and the fission product release is subsequently increased.

Some failed fuel was located in areas of the core considerably removed from the high probability suspect areas defined by flux tilting. These bundles were discovered as the result of sipping bundles that were removed for other purposes, such as LPRM replacement or CRD maintenance. Sufficient time was not available during this outage schedule to sip the entire core. Therefore, it is probable that some leaking bundles remain in the core and there will be some offgas release during subsequent operation.

The 29 failed fuel assemblies will be disassembled and each rod examined and tested individually, and the assemblies will be reconstituted for subsequent re-use. The technique of examination, reassembly and re-use of fuel has been demonstrated satisfactorily at KRB and Big Rock Point. Plans are underway to return typical failed rods to the Radioactive Materials Laboratory to verify the postulated failure mechanism.

It is believed that the core as presently reconstituted is in a safe and operable condition. It is quite probable that most of the faulty fuel has been identified and removed in this early rash of failures. In any event, the restart of the reactor will be conducted in a careful stepwise escalation of power, with careful monitoring of the condition of offgas and reactor water.

DRESDEN II

CONTROL ROD DRIVE OPERATIONAL REPORT

JULY 29, 1970

DRESDEN-2 CONTROL ROD DRIVE OPERATIONAL REPORT

During the Dresden 2 startup test program, it was found on several occasions that several control rod drives were experiencing an increase in scram time between 0.4 and 1.8 seconds. Just prior to the June 5 incident (May 29th thru May 31) tests were conducted to determine scram time performance of all control rod drives. These tests revealed a scram time distribution as indicated on Fig. 1.

In addition, some drives were identified as having high stall flows (In excess of 5 GPM).

As a result of these two conditions the following action was taken during the outage following the June 5 incident:

1. All available data taken during in-reactor testing was evaluated.
2. "Plugged filter" simulation tests were conducted in San Jose.
3. Disassembly and inspection of selected drives.
4. Stall flow tests were conducted on selected drives.

Six (6) drives were removed for inspection. Four (4) of these were drives having scram times greater than 10 seconds and two (2) were drives having stall flows greater than 5 GPM.

The following CRD's were disassembled and inspected:

#456C	(Location 26-51)	slow scram
#292	(Location 30-31)	slow scram
#688C	(Location 34-27)	slow scram
*#619C	(Location 14-11)	slow scram (0.002" filter)
#118	(Location 14-35)	high stall flow
#605C	(Location 34-31)	high stall flow

* Drive 619C had been considered inoperative prior to the in-reactor tests.

In-Reactor Testing

In-reactor testing was conducted and verified that no abnormality existed outside the control rod drive assemblies, which would cause slow scram times.

San Jose Simulation

Reactor scram conditions were simulated in San Jose tests with plugged inner screen and mechanical clearances which produced equivalent drive performance to the in-reactor tests.

Inspection of CRD's

All six drives had a heavy accumulation of reddish fines "goo" on the top of

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the stop piston and the I.D. of the index tube. Inspection of the four slow scram drives revealed severely bulged end caps on the inner screen. There was no indication of bulging on the two high stall drives. Further, except for heavy amounts of foreign material in the stop piston area, no other control rod drive abnormality was observed.

Evaluation of the data collected from in-reactor testing, San Jose simulation, and inspection of the drives indicate that the cause of excessive scram times can be attributed to foreign material plugging the control rod drive inner screen. This plugging acts to exert an additional force in opposition to the force for causing scram motion when at rated pressure and temperature.

Stall Flow Tests

Testing was divided into two groups:

1. Drives having reported stall flows greater than 4.5 GPM.
2. Drives having zero stall flows.

These tests did not indicate a condition of high stall flows and the validity of the stall flow data previously collected is doubtful. Suspicion is directed at the accuracy and reliability of the flow measurement instrumentation. This instrumentation will be thoroughly checked out and calibration accuracy verified. Once the instrumentation is considered satisfactory, stall flow tests will be repeated.

In summary, all tests and inspections to date indicate that the slow scram times are caused by plugging of the inner filters and are not attributed to the drive assembly or hydraulic system design. Corrective action taken thus far has been to replace the six (6) drives which were removed. Drive performance will continue to be observed during forthcoming power operation.

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