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RD 9/27

SEP 27 1974

Docket No. 50-237

Commonwealth Edison Company
ATTN: Mr. J. S. Abel
Nuclear Licensing Administrator - Boiling Water Reactors
P. O. Box 767
Chicago, Illinois 60690

Gentlemen:

The Regulatory staff has completed its review of the proposed repair program for the Dresden Station Unit 2 recirculation line as described in your report entitled "Commonwealth Edison Company Dresden Station 2A Recirculation Pump 4" Equalizing Line Repair Program." This report was transmitted by your letter dated September 23, 1974.

Based on our review, we have concluded that the proposed repair program can be performed as described with reasonable assurance that the health and safety of the public will not be endangered and therefore is acceptable. The enclosed Amendment No. 3 to Facility License No. DPR-19 incorporates in the license authority for you to make the proposed repairs.

A copy of our related Safety Evaluation and Federal Register Notice also are enclosed.

SVarga
RSchemel
ACRS (16)
HMcAdluff, ORO
JRBuchanan, ORNL
TBAbernathy, DTIE

Sincerely,

Original signed by:
Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Amendment No. 3 to DPR-19
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

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SURNAME ➤	RSilver:aw	DLZiemann			KRGoller	
DATE ➤	9/27/74	9/27/74	9/27/74	9/27/74	9/27/74	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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of Grundy County
Grundy County Courthouse
Morris, Illinois 60450

cc w/encls. and filing dtd.
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Illinois Department of Public Health
Springfield, Illinois 62706

Mr. Gary Williams
Federal Activities Branch
Environmental Protection Agency
1 N. Wacker Drive, Room 822
Chicago, Illinois 60606

22462

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COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

(DRESDEN UNIT 2)

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. DPR-19

1. The Atomic Energy Commission (the Commission) has found that:

   A. The filing by the Commonwealth Edison Company (the licensee) dated September 23, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public, and

   E. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, Facility License No. DPR-19 is hereby amended to add paragraph 3.E to read as follows:

   3.E For the purpose of repairing a crack in the recirculation bypass line in the "A" loop, the licensee may perform the repair program as described in a report entitled "Commonwealth Edison Company Dresden Station 2A Recirculation Pump 4" Equalizing Line Repair Program" transmitted by letter dated September 23, 1974.

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3. This license amendment is effective as of the date of its issuance.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by:
Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Date of Issuance: SEP 27 1974

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SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

DOCKET NO. 50-237

COMMONWEALTH EDISON COMPANY

I. INTRODUCTION

On September 12, 1974, the Commonwealth Edison Company shutdown Dresden Unit 2 upon detection of abnormally high primary coolant leakage. On September 13, they found a circumferential crack at a weld in the four inch diameter schedule 80 stainless bypass line around the "B" loop recirculation pump discharge valve. This crack was on the pipe side of the weldolet located between the recirculation pump and the pump discharge valve. Since this location could be isolated from the reactor vessel by closing the valves in the bypass and recirculation lines the crack was subsequently repaired using routine procedures.

On September 15, 1974, inspection of the 4" diameter bypass line on the "A" loop disclosed a similar crack but downstream of the recirculation pump discharge valve so that it cannot be isolated by system valves. The Commonwealth Edison Company (CE) has therefore designed and fabricated special fixtures and developed special procedures for effecting repairs of the "A" loop bypass line. By letter dated September 23, 1974, CE requested AEC approval of a proposed repair program. The program involves the use of temporary plugs to avoid draining of reactor vessel water for the repair which would otherwise necessitate the very time consuming operation of first unloading all the fuel from the reactor vessel. The CE report describing the repair program includes a description and evaluation of the repair procedure, the back up measures to prevent draining reactor vessel water if a temporary plug should fail and a safety evaluation of a postulated failure of a plug and the back up measures.

II. PROPOSED REPAIR PROCEDURE

The applicant's proposed procedure for isolating the area where the replacement pipe will be welded involves the use of temporary freeze plugs and an expandable rubber plug. Back up plugging techniques to stop the loss of coolant that would occur in the event of failure of a temporary plug have been developed and demonstrated.

As stated in the licensee's report, Commonwealth Edison has acquired considerable experience with the use of freeze plugs. Through the use of two full scale mock-ups, they have developed procedures, functionally verified the repair apparatus and back up devices, and have trained their repair crews.

A freeze plug/welding mock-up has been used to verify the freeze-plug procedure and its compatibility with welding in close proximity to the freeze plug. Freeze plugs have been maintained in excess of two hours after shut-off of the refrigerator without any evidence of leakage.

A hydraulic mock-up has been used to verify the satisfactory operation of the back-up sealing devices as well as the conduct of the entire repair procedure.

An AEC Regulatory Operations inspector has witnessed numerous demonstrations of the freeze plug technique using these mock-ups and indicates that the licensee's personnel who have been trained to perform this function have become quite proficient in the technique and can consistently provide freeze plugs with no leakage or freeze plug movement.

All the operations that will be required of the rubber plug-rod-stuffing box assembly and Plidco clamp, which is a commercially available item, were evolved and qualified in runs made on two full-scale mock-ups. All tests were performed by the licensee. A number of simulation runs on the mock-up have demonstrated the reliability of the system under realistic conditions. Tests to determine the heat resistance of the rubber material used for the plug indicate that it can provide good sealing at temperatures as high as 500°F. Exact procedures for the operation of the system have been worked out with the use of the mock-up. An integral part of this procedure are back-up methods for sealing in case any leakage should occur.

We concur with the licensee that the Rubber Plug-Rod Stuffing Box-Cap Assembly and Associated Plidco Clamp can be relied upon to safely perform its intended function of stopping the flow of coolant. This opinion is based on our review of the applicants test program, and the observations of the AEC Directorate of Regulatory Operations inspector who has witnessed testing of this plug.

A crew of six experienced welders will be available for the welding repairs; all have been qualified in accordance with ASME Code Section IX. The welding procedures to be employed for the repair have similarly been qualified in accordance with the requirements of Section IX. Nondestructive examinations of sample welds exceeded code requirements. Radiographic examination and liquid penetrant examination will be made after the second welding pass and after the final pass, although the applicable code requires only liquid penetrant examination during the second pass and liquid penetrant and radiographic examination after the final pass.

Precautions have been taken to ensure that the expandable rubber plug will not be damaged by heat from the welding operation. Tempil sticks will be used to monitor the temperature where the expandable rubber plug is located, approximately 1 inch from the weld. Welding will be stopped whenever the indicated temperature reaches 250°F. Since tests run by the applicant have indicated that the expandable rubber plug can provide good sealing at temperatures up to 500°F, a considerable margin of safety will be present during the welding operation.

III. EVALUATION OF ALTERNATIVE COURSES OF ACTION

In the course of evaluating corrective actions for the "A" loop crack, CE considered several alternative repair programs. These included (1) unloading the core and draining the reactor vessel, (2) removing the reactor vessel head, dryer and separator and plugging the jet pumps, (3) using a freeze plug in the 28 inch diameter line and (4) using freeze plugs in the 5-twelve inch diameter jet pump risers.

The first alternative would involve essentially no risk of draining water from the reactor vessel but would involve other safety considerations resulting from extensive handling of irradiated fuel and higher occupational exposures than the proposed program. We evaluated the occupational radiation exposures that would be expected as a result of following this alternative repair method. Based on a CE letter to D. J. Skovholt, dated April 18, 1974, the Dresden plant averaged 67 man-rem for refueling operations in 1973. Considering that a full core would have to be unloaded, instead of one third core as is typical, and reloading of irradiated fuel instead of new fuel, we conclude that a total occupational radiation exposure of 200-300 man-rem may be involved. This compares with a total exposure at the Dresden plant of about 900 man-rem for 1973. By comparison, the estimated occupational exposure that will occur during the proposed repair procedure is less than 100 man-rem. From an occupational exposure viewpoint, this alternative method would not be desirable as the repair method proposed.

Alternative 2, plugging of the jet pumps, was not selected because the design, fabrication and testing of the necessary special tools has not been completed. Furthermore, this alternative would also have the potential for failure of temporary plugs, which would have the same result as for the proposed repair procedure.

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Alternative 3, using freeze plug techniques on the 28" diameter recirculation pump discharge line is not feasible, because at present, even under laboratory conditions, 18" diameter is the largest pipe upon which the freeze plug technique has been demonstrated.

Alternative 4, would involve establishing a freeze plug in each of the five 12" diameter jet pump riser lines from the manifold leading to the jet pumps. The use of freeze plugs on the five lines would permit drainage of the recirculation line and the cracked bypass line. However, at present, Commonwealth Edison does not have a proven freeze plug procedure for a 12-inch diameter pipe. In addition, this procedure would require freezing five plugs in parallel flowpaths, so that natural circulation flow would inhibit freeze plug formation. Since the applicant does not currently have confidence in his ability to freeze 12" lines, we concur that this alternative course of action should not be considered.

IV. POTENTIAL CONSEQUENCES OF PLUG FAILURE

During the repair, failure of both the primary and backup plugging techniques would result in a loss of coolant through a four-inch pipe. The probability of this occurring during the repair is very small, but it is higher than an equivalent size failure of the reactor coolant system boundary occurring during normal operation. However, the consequences of a loss of coolant through a 4" pipe under conditions when the reactor has been shutdown and depressurized for two weeks and primary coolant temperature has been reduced to about 100 to 130°F are minimal and are insignificant compared with the consequences of the loss-of-coolant accidents postulated to occur during normal operation as evaluated in the FSAR. Since the break would be small and the system pressure low, the flow from the condensate system or any ECCS pump would be sufficient to maintain normal water level. If a temporary plug failed, the flow out of the four-inch pipe would be approximately 2500 gpm. The flow from a core spray pump would be at least 4500 gpm and the flow from a LPCI pump would be at least 5000 gpm. Operability of the LPCI pumps and the onsite power supply will be demonstrated prior to commencement of the repair to assure that a supply of makeup water would be available. The preferred source of makeup water from the condensate system could be maintained for about 40 minutes and the injection of water by the ECCS pumps could be maintained indefinitely since these take suction from the primary containment suppression pool, the ultimate sump for the water leaving the break. The redundant residual heat removal heat exchangers and the service water system would be utilized to remove the fission product decay heat.

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Since the temperature of the fuel and clad are nearly the same as the 100 to 130°F temperature of the reactor coolant and the decay heat generation from fission products is low, forced convection cooling is not required and the fuel and clad temperatures will not increase as long as the core remains covered with coolant. Therefore, fuel damage and a consequent release of fission products would not occur even if a temporary plug failed during the repair.

The staff estimates that the only gaseous radioactive release from plug failure would be that of I-131 at a rate of 5.5 x 10^{-5} Ci/hr after being processed by the Standby Gas Treatment System and released from the plant stack. The maximum release time postulated for this release path is 24 hours or a total I-131 release of 1320 uCi. Assuming the 5 percentile accident meteorology for the Dresden site of 2.8 x 10^{-6} sec/m^3 for the atmospheric dispersion factor of a stack release, the staff calculates a peak ground level concentration of 4.3 x 10^{-14} uCi/cm^3 beyond the site boundary.

The potential thyroid dose for a release of 1320 uCi of I-131 under accident meteorology to an individual at the nearest site boundary would be 0.0013 mrem by inhalation. The potential thyroid dose to a child by way of the milk pathway due to such a release would be 0.31 mrem. Such potential thyroid doses are significantly less than 1 percent of the non-occupational annual dose limits specified in 10 CFR Part 20 and are more than a factor of 10 less than the design levels necessary to meet as low as practicable annual thyroid doses of 5 mrem. It is expected that in case of such an incident, meteorological and milk measurements would be carried out promptly to determine the exact extent of possible consequences to individuals offsite. For release periods beyond the postulated 24 hours, the expected releases would be those associated with a normal refueling outage.

The staff also evaluated potential doses to workers in the drywell in the event of a plug failure. Our calculations are based on the isotopic composition and radioactivity level which the licensee reports currently exists in the reactor coolant water and an exposure time of 15 minutes established by the licensee as a maximum time to be spent by the repair crew in attempting to establish a back up seal if a primary plug fails. Our calculations indicate a potential whole body exposure of 2.1 mrem and a skin exposure of 2.5 mrem for the 15 minute period. A whole body ingestion dose of 1 mrem and a thyroid ingestion dose of 30 mrem could result from a postulated ingestion of 100 cm^3 of reactor coolant water. These potential exposures to personnel from a plug failure, the staff concludes that the potential exposures

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represent less than 1 percent of the annual occupational exposure limits specified in 10 CFR Part 20. Even with significantly longer exposure periods, the doses to the workers would be within the exposure levels of Part 20.

V. CONCLUSIONS

As discussed above, the staff considers the proposed repair plan acceptable and the best of the feasible alternatives. The chance of failure of the temporary plugs is very small, but is judged to be higher than that of an equivalent size break occurring in the primary coolant system boundary. However, the energy which has to be removed from the core is a small fraction of the energy that would have to be removed following a loss of coolant accident. In addition, the temperature and initial pressure are much lower than for situations analyzed in the FSAR. Therefore, the time available to take corrective action is greater and the heat removal capacity needed is very much less than would be required for a loss of coolant accident involving the same pipe under operating conditions. Adequate makeup coolant to keep the core covered and maintain fuel element temperatures near the ambient temperature of 120°F can be provided by any one of the four low pressure coolant injection pumps or the two core spray pumps. Accordingly, the potential consequences of such a failure of the plugs in the cold reactor which has been shutdown for two weeks would not result in cladding damage or release of fission products from the core. The only radiological consequences would be those described above, which are insignificant compared to an equivalent size break in the piping of a hot pressurized reactor at power which was analyzed in the Final Safety Analysis Report as amended (FSAR).

Having analyzed the potential sequence of events, including consideration of a number of malfunctions, we conclude that even if the temporary plugs fail completely during the short period that the four inch pipe is completely severed, there will be, at all times, a sufficient water supply to prevent damage to the core. We also conclude there is reasonable assurance that no significant amounts of fission products would be released from the core. If the plugs and back up devices should fail and the repair crew is not able to establish a seal within 15 minutes, the licensee will evaluate the situation and if a program for stopping the leakage is not completed within 24 hours, the licensee will remove the reactor vessel head, unload the core and complete the repair without fuel in the core.

The only potential radiological consequences that need to be considered are those that would result from working in a radiation field and from the release of radioactivity from the primary coolant that would drain into the containment. As discussed above, both the occupational and offsite doses would be well below the limits of 10 CFR Part 20. In regard to radiological consequences to personnel, we conclude that the maximum potential consequences of the proposed repair program are not significantly different from those of normal maintenance activities. The maximum offsite radiological effects will not be significantly different than from routine operation.

Based on the above considerations, we have concluded the proposed repair procedure does not involve a significant hazards consideration. We also conclude that there is reasonable assurance (i) that the proposed activities can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations and will not be inimical to the common defense and security or to the health and safety of the public.

Richard D. Silver
Operating Reactors Branch #2
Directorate of Licensing

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Date: SEP 27 1974

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UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-237

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY LICENSE

Notice is hereby given that the U. S. Atomic Energy Commission (the Commission) has issued Amendment No. 3 to Facility Operating License No. DPR-19 to the Commonwealth Edison Company (the licensee). The license authorizes operation of the Dresden Nuclear Power Station Unit 2 (the facility) located in Grundy County, Illinois.

The amendment authorizes the licensee to repair a crack in the recirculation bypass line in the "A" loop of the facility in the manner described in a report entitled "Commonwealth Edison Company Dresden Station 2A Recirculation Pump 4" Equalizing Line Repair Program" transmitted by letter dated September 23, 1974. The amendment is effective as of its date of issuance.

The filing dated September 23, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I which are set forth in the license amendment.

For further details with respect to this action, see (1) the licensee's filing dated September 23, 1974, (2) Amendment No. 3 to License No. DPR-19, and (3) the concurrently issued related Safety Evaluation by the Regulatory staff of the Commission. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Morris Public Library at 604 Liberty Street in Morris, Illinois 60451. A single copy of items (2) and (3) may be obtained upon request addressed to the Atomic Energy Commission, Washington, D.C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland this 27th day of September, 1974.

FOR THE ATOMIC ENERGY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing