MAR 15 1974

Docket No. 50-249

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

Gentlemen:

A copy of the Staff Safety Evaluation relating to your request dated September 14, 1973, as amended with Supplements A through H thereto, for the use of 8 x 8 fuel assemblies at the Dresden Nuclear Power Station Unit 3 is enclosed for your information. The evaluation reflects the Staff's review of the expected performance of General Electric 8 x 8 fuel bundles in the Dresden Nuclear Power Station Unit 3, the effects of fuel densification on the performance of 8 x 8 and 7 x 7 fuel assemblies, proposed changes to the Technical Specifications, and other aspects of operation with Reload 2. Action regarding authorization of operation with the reload fuel and proposed changes to the Technical Specifications is awaiting expiration of the 30-day notice period.

Sincerely,

Original Signed by Robert J. Schemel
[redacted]

for Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosure:
Safety Evaluation

cc: See next page

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UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

INTRODUCTION

By application dated September 14, 1973, Commonwealth Edison Company (CE) requested authorization to operate Dresden Unit 3 with Reload 2 fuel assemblies in the core. According to CE's plan, approximately 68 Reload 2 fuel assemblies will replace an equal number of assemblies presently in the core. Reload 2 is to consist of approximately twenty-four 7 x 7 fuel assemblies similar to fuel presently in the core and forty-four 8 x 8 fuel assemblies. In addition, the remaining 140 temporary fixed poison curtains placed in the original loading for supplementary reactivity control will be removed. The application also includes a request for approval of proposed Technical Specifications related to fuel densification considerations and the rod block monitor.

Supplements to the application were submitted by letters dated November 27, December 6 (2 letters), December 17, 1973 (2 letters), January 9 (2 letters), January 18, and January 23, 1974.

The safety analysis of the reload submittal by the licensee includes evaluation of the effect of the reload on previously analyzed conditions during normal operation, operational transients, and postulated accidents. Included in this analysis is consideration of applicability of existing Technical Specification operating limits and the evaluation of proposed revisions to limits. The evaluation included consideration of the reload fuel bundles of the presently used 7 x 7 array, the new design reload fuel assemblies in an 8 x 8 array, and the characteristics of the reactor with a combination of the initial fuel assemblies and reload fuel assemblies. The acceptability of the neutronic, thermal-hydraulic, and mechanical design of the 8 x 8 assemblies during normal operation, operational transients, and postulated accidents is evaluated by the

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Regulatory staff in a previous report[1]. The use of 8 x 8 fuel for reloads was also reviewed by the Advisory Committee on Reactor Safeguards and discussed in its report dated February 12, 1974.

The 7 x 7 Reload 2 fuel design is the same as the Reload 1 fuel presently in the core. The applicant's methods of analysis, used and approved for previous loads, are therefore applicable to the 7 x 7 Reload 2 fuel.

EVALUATION

The reference core consists of 604 initial 7 x 7 fuel assemblies, fifty-two 7 x 7 Reload 1 assemblies, twenty-four 7 x 7 Reload 2 assemblies which are identical to Reload 1 assemblies, and forty-four 8 x 8 assemblies. The reload assemblies will be in a symmetric one-reload-assembly-in-four-assembly type array. No significant fuel loading asymmetries will exist. Therefore, the fuel types and loading pattern fall within the scope of the staff report on the 8 x 8 fuel assemblies[1]. The thermal-hydraulic limits and the response of the coolant circulation system with jet pumps are the same as that evaluated in the staff report. The methods of analysis used by the licensee are identical to the methods approved by the staff. Therefore, the evaluations and conclusions of the staff report with respect to normal operations, abnormal operational transients, and accidents are fully applicable to Dresden 3.

The Regulatory staff's review[1] of the mechanical design of the 8 x 8 reload fuel concludes that the background of experience compiled by the General Electric Company is sufficient to enable GE to design fuel rods of new design with confidence in their durability. The Dresden 3 8 x 8 fuel assemblies are of similar design and material to the 7 x 7 fuel assemblies which have successfully been operated at Dresden 3. Both the 8 x 8 and 7 x 7 assemblies will operate at the same pressure and temperature and the fluid velocity and quality will be nearly identical and, therefore, the new 8 x 8 fuel assemblies are expected to exhibit the same operational characteristics as the previously operated 7 x 7 assemblies. In addition, an out-of-pile flow test of a similar prototype bundle provides further assurance of the adequacy of the design. A surveillance program, to monitor the performance of the new fuel assembly design, will also be performed.

Accident induced loads and stresses have been calculated for both the 7 x 7 and 8 x 8 assemblies using the same methods. The limiting accident loads result from a steam line break. The pressure differences following

(1) "Technical Report on the General Electric Company 8 x 8 Fuel Assembly" dated February 5, 1974, by the Directorate of Licensing.

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a steam line break are less than 10% greater than normal operating pressure differences. As in normal operation, the pressure differences in an 8 x 8 assembly following a steam line break are 5 to 10% greater than in a 7 x 7 assembly. The loads following a steam line break are well below the allowable loads.

Based upon the above, the staff concludes that the mechanical design of the Dresden 3 8 x 8 reload fuel is adequate to assure the mechanical integrity of the fuel assemblies. Additional assurance of acceptable fuel performance of the new fuel is provided by the radiological surveillance performed on the reactor primary coolant and off-gas to provide an early indication of incipient fuel failure caused by mechanical deterioration of the fuel assemblies.

We have also reviewed the nuclear design of the 8 x 8 reload fuel. The fuel is identical to that which is evaluated in the Regulatory staff's evaluation(1) of 8 x 8 fuel elements. We conclude that a mixed 8 x 8 and 7 x 7 core will be nearly identical, neutronically, to a 7 x 7 core and that the nuclear design is acceptable.

The staff evaluation(1) of the expected thermal-hydraulic performance used identical fuel damage limits and thermal-hydraulic criteria to evaluate both the 8 x 8 and 7 x 7 assemblies. The results of this evaluation show that the 8 x 8 assembly minimum critical heat flux ratio (MCHFR) is expected to be 11% greater than the MCHFR for a 7 x 7 assembly operating at the same power. Additionally, the 8 x 8 fuel assemblies operating at their design LHGR value have a 20% greater margin to the 1% cladding strain criterion than the 7 x 7 assemblies and the margin of design linear heat generation rate to pellet center line melting is 17% higher for 8 x 8 assemblies than for 7 x 7 assemblies. The staff has also reviewed the basic hydraulic differences between the 7 x 7 and 8 x 8 assemblies which are the modified flow geometry and the introduction of an unfueled rod. The modified flow geometry will provide a more balanced subchannel flow in the 8 x 8 assembly than in the 7 x 7 bundle and therefore we conclude that the thermal performance is improved. The effect of the unheated rod has been previously reviewed(2) and the staff concluded that the effect of the unheated rod is not significant.

Based on the above considerations, the staff concludes that the thermal-hydraulic performance of the Dresden 8 x 8 reload fuel is acceptable and will provide an increased margin of safety as compared with the previously operated 7 x 7 assemblies.

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(2) Change No. 17 for Oyster Creek, Docket No. 50-219, License No. DPR-16, letter from D. J. Skovholt to Ivan Finfrock, Jersey Central Power and Light Company, dated November 16, 1973.

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A. Proposed Changes to Technical Specifications

Since the performance characteristics of the Reload 2 bundles are similar to the previously authorized loadings, the safety limits and limiting safety system settings presently specified in the Technical Specifications are applicable. With two exceptions, the limiting conditions of operation also are unchanged. These exceptions are the upscale level trip set point of the Rod Block Monitor (RBM) and the average planar and local linear heat generation rate (LHGR) limits.

The rod block monitor provides protection of the core in the event of an inadvertent withdrawal of a control rod of high reactivity worth from a limiting control rod pattern. Specifically, the rod block provides local protection of the core by limiting control rod withdrawal so that the minimum critical heat flux ratio (MCHFR) is maintained above 1.0. At present, the upscale trip level set point of the RBM is specified to be equal to or less than 0.65 W + 45 where W is percent of design recirculation flow. At this setting, rod withdrawal is blocked when MCHFR is about 1.6. A reanalysis performed by the applicant for the reloaded core shows that the upscale trip level set point has to be reduced to provide a margin in MCHFR above 1.0. The change is not required by the use of the 8 x 8 fuel. Commonwealth has proposed to reduce the limiting set point to 0.65 W + 42. At this reduced setting the reanalysis shows that the rod withdrawal is blocked when MCHFR is about 1.16. Although the margin to a MCHFR of unity is reduced, it is still within the range of approved values for other reactors and provides a margin which we consider acceptable. The proposed setting reduction to a more restrictive value is therefore acceptable.

Average planar and local LHGR is a function of the fuel type and is related to fuel densification. Since a new fuel type (the 8 x 8) is being added to the core, new limitations must be incorporated in the Technical Specifications. In addition, the limits for the 7 x 7 fuel can be modified as a result of a revised fuel densification model approved by the staff in December 1973(3). The use of the revised densification model and the resultant change in average planar and local LHGR limits was reviewed for Dresden 2 and approved by Change 24 to DPR-19 dated December 28, 1973. That evaluation concluded that the use of the new model has essentially

(3) Technical Report on Densification of General Electric Reactor Fuels Supplement 1, December 14, 1973, USAEC Regulatory staff.

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no effect on normal operation and improves the margins to pressure and minimum critical heat flux ratio limits for overpressurization and core flow reduction transients. The staff also concluded that the limitations of the average linear heat generation rate of all rods in any fuel assembly, calculated by use of the new fuel densification model, assure that the calculated peak clad temperatures in the event of a loss-of-coolant accident will not exceed 2300°F. The staff report on 8 x 8 fuel(1) notes that the fuel densification model is equally applicable to the 8 x 8 fuel. Therefore, the proposed technical specifications for average planar and linear LHGRs, calculated by use of the approved fuel densification model, are acceptable.

B. <u>Removal of Poison Curtains</u>

The temporary fixed poison curtains were inserted into the original loading for supplementary reactivity control. The need for this supplementary poison generally decreases with increasing core exposure. The applicant has calculated the shutdown margin for the next fuel cycle. Their calculations show that the minimum shutdown margin with all the temporary poison curtains removed is well in excess of that required by the Technical Specifications. The minimum shutdown margin will be verified by measurement prior to startup. Therefore, we concur with the licensee's plan to remove all the temporary poison curtains prior to operation with the Reload 2 core.

C. <u>Abnormal Operational Transients</u>

Abnormal operational transients were discussed in the staff report for 8 x 8 reload fuel(1). As previously discussed, the mechanical, nuclear, and thermal-hydraulic characteristics of the 7 x 7 and 8 x 8 fuel are similar and will respond to transients similarly. Also, the reduction in flow in the 8 x 8 assemblies will be offset by an accompanying flow increase in the 7 x 7 assemblies and the effect on the total core flow will be negligible.

The staff also concludes that the replacement of the 7 x 7 assemblies with 8 x 8 assemblies will not result in exceeding fuel damage limits during anticipated transients. The licensee has analyzed the events which have limiting MCHFRs, including a seizure of one recirculation pump, a continuous withdrawal of a control rod, and misorientation of a fuel assembly. The results show that the fuel damage limit, a MCHFR of unity, is not reached during these transients. However, one postulated operational transient, the turbine trip without bypass, necessitated a steady state power reduction in the last cycle to acceptably limit the calculated primary system pressure

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increase. The power reduction requirement resulted from a reanalysis with revised control rod scram reactivity curves. The pressure increase and scram reactivity analysis is primarily a function of core exposure (reactivity) and is applicable to the combined loading of 8 x 8 and 7 x 7 fuels. Commonwealth estimates that at approximately four months into the cycle, the maximum steady state power will have to be limited to 97% power and five months later the power would have to be further reduced. Limiting power to assure the maintenance of present pressure safety margins during the postulated turbine trip without bypass transient is acceptable. The staff will require Commonwealth to submit another scram reactivity analysis for our approval prior to the nine-month interval in which the existing analysis applies.

D. Accident Analysis

The generic reevaluation of accidents to account for the effects of 8 x 8 fuel was discussed in the staff evaluation(1) and is applicable to Dresden 3. That evaluation noted that the plant specific aspects of the review, such as compliance with the Interim Acceptance Criteria for Emergency Core Cooling, including the effects of densification, any necessary revisions to Technical Specifications requirements, and radiological consequences of postulated accidents would be addressed in the separate evaluation for the specific plant. The Technical Specifications changes, including those associated with densification, have been discussed above.

The Regulatory staff has reviewed the analysis of the loss-of-coolant accident presented by Commonwealth Edison and has concluded that the General Electric Evaluation Model (NEDO-10329), as modified by GE in NEDE-10801 to account for differences in geometry and subsequently modified by the staff to account for the effects of fuel densification, is applicable to the evaluation of the Emergency Core Cooling performance of 7 x 7 assemblies. The staff further has concluded(1) that this model is also applicable to the evaluation of 8 x 8 fuel assemblies in a General Electric boiling water reactor which has jet pumps. The result of the application of these approved General Electric fuel densification evaluation models to predict the specific ECCS performance at Dresden Unit 3 operating in accordance with proposed Technical Specifications shows that the peak clad temperature for the 7 x 7 initial loading, 7 x 7 reload, and the 8 x 8 reload fuel remains below 2300°F, and that the metal water reaction is less than one percent, thereby meeting the requirements of the Interim Acceptance Criteria for Emergency Core Cooling.

The radiological consequences of the postulated accidents is a function of the fission product release, including any change in fission product release because of the use of 8 x 8 fuel. The radiological consequences of a steam line break, fuel handling, rod drop, and loss-of-coolant accidents were considered. As noted in the staff 8 x 8 report, the steam line break accident is almost entirely dependent on the limits placed on concentration of radioactivity in the primary coolant. These limits are not being modified and therefore the radiological consequences remain essentially unchanged. The resulting radiological doses will remain under ten percent of 10 CFR Part 100 guidelines.

The fuel handling accident is dependent on the damage resulting from dropping an irradiated fuel element on other fuel elements. Since an 8 x 8 bundle is the same size and approximately the same weight as a 7 x 7 bundle, it would impart the same energy to the same number of fuel assemblies as a dropped 7 x 7. Since the 8 x 8 fuel assembly design and fission product inventory are similar to the 7 x 7, the radiological consequences of dropping an assembly onto an 8 x 8 assembly will not be significantly different. The doses from a refueling accident are calculated to be less than ten percent of 10 CFR Part 100 guidelines. Analyses of the rod drop accident demonstrate that the dropping of a maximum worth sequenced control rod will not result in a peak fuel pellet enthalpy which exceeds the present limit of 280 calories/gram. The number of 8 x 8 rods in the core which would perforate as a result of such an energy deposition is estimated to be higher than the number of 7 x 7 rods which would perforate as a result of a rod drop accident. However, the radiological consequences would be nearly the same because rod power is lower in the 8 x 8 rods. The calculated design basis loss-of-coolant accident doses are based on a conservatively large fission product inventory release which is independent of the calculated number of perforations which would occur during a LOCA and the release through perforations. Therefore, the calculated radiological doses from the design basis loss-of-coolant accident would also remain unchanged by the use of 8 x 8 fuel assemblies.

CONCLUSION

Based on the above, we have concluded that the health and safety of the public will not be endangered by the proposed refueling and subsequent operation with Reload 2 and with the proposed modifications to the Technical Specifications.

/s/

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

/s/

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

Date: MAR 15 1974

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