MAR 25 1974

Docket Nos. 50-254 and 50-265

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

Change No. 13
Licenses Nos. DPR-29
and DPR-30

Gentlemen:

Your letter dated November 16, 1973, submitted proposed changes to the Technical Specifications of Quad-Cities Units 1 and 2. These changes concern maximum allowable in-sequence control rod worth and control rod scram insertion time requirements in Section 3.3 of the Technical Specifications and associated bases in Section 1.2, 2.1 and 3.3.

We have completed our review of the proposed changes based on your submittal and the referenced documentation. Based on this review, we have found that the maximum in-sequence control rod worth may be increased from 1% delta k to 1.3% delta k and that the allowable control rod scram rates should be reduced to the insertion time stated in your submittals. In addition, we have revised Specification 3.3.B.3.b relating to the Rod Worth Minimizer, referred to in item 3 of your letter dated November 6, 1972. We have also made minor additional changes to the Technical Specification Section 3.3 which were discussed with the staff of Commonwealth Edison Company.

We have concluded that the above changes in Section 3.3 of the Quad-Cities Technical Specifications and associated bases in Sections 1.2, 2.1 and 3.3, as modified by us, do not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the manner accorded by these changes.

MAR 25 1974

Pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications and Bases of Facility Licenses Nos. DPR-29 and DPR-30 are hereby changed by replacing the present pages 16, 25, 74, 75, 76 and 82 with the enclosed revised pages and additional pages 82A and 82B.

These changes to the Quad-Cities Technical Specifications shall become effective the date of this letter, except Section 3.3.B.3.b which will become effective not later than 6 months from the date of this letter.

A copy of our Safety Evaluation relating to these Technical Specifications changes is also enclosed.

Sincerely,

/S/

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

1. Revised pages
2. Safety Evaluation

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Form AEC-318 (Rev. 9-53) AECM 0240

2.1 Limiting Safety System Setting Bases (cont'd)

the scram worth of about 75% of the control rods. The scram delay time and rate of rod insertion are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The insertion of the first dollar of reactivity strongly turns the transient and the
13 stated 5% and 20% insertion time conservatively accomplishes this desired initial effect. The time for 50% & 90% insertion are given to assure proper completion of the insertion stroke, to further assure the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

The design peaking factors at the full power conditions for Quad-Cities result in a MCHFR value of 2.04. For analysis of the thermal consequences of the transients, higher peaking factors are used, such that a MCHFR of 1.9 is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the rated power level produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels. Sensitivity analyses, referenced in the SAR, Amendment 11, Section 4, Question 2.1.2
4 indicate that for the turbine trip with flux scram without bypass or relief, a significant reduction in the neutron flux and heat flux peaks will be realized when the smaller void reactivity coefficient is used. For this particular transient, if it were also analyzed at a power level of 110% of rated but with the expected void reactivity coefficient, the resulting heat flux peak would be less than the peak resulting from the analysis actually conducted from rated power but with the conservative void coefficient.

Inherent in these analyses is the fact that steady-state operation without forced recirculation flow will not be permitted except during startup testing.

In summary, the transients presented in the SAR were analyzed only up to the design flow control line (see Figure A) and not above because:

1. The licensed maximum steady-state power level is 2511 MW(t).

2. The units cannot physically be brought above 2511 MW(t) unless abnormal operation is employed.

3. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

4. The analysis model itself is demonstrated to be conservative.

5. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power, which has been shown above to be unrealistic, than using conservative values for the parameters.

1.2 Safety Limit Bases

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, and coolant system piping. The respective design pressures are 1250 psig at 575°F, and 1175 psig at 560°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel
3 and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% X 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the primary coolant system.

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield
3 strength are comparable for the primary system piping and provide a similar margin of protection at the established safety pressure limit.

The normal operating pressure of the reactor coolant system is 1000 psig. For the turbine trip or loss of electrical load transients, the turbine trip scram or generator load rejection scram, together with the turbine bypass system limit the pressure to approximately 1100 psig(4). In addition, pressure relief valves have been provided to reduce the probability of the safety valves operating in the event that the turbine bypass should fail. These valves and the neutron flux scram
13 limit the reactor pressure to 1185 psig(5,6,7) which is 25 psi below the setting of the first safety valve. Finally, the safety valves are sized to keep the reactor coolant system pressure below 1375 psig with no credit taken for the relief valves or turbine bypass system. Credit is taken for the neutron flux scram, however.

(4) SAR Section 11.2.2.
(5) SAR Section 4.4.3.
(6) Dresden 3 Special Report No. 29, "Transient Analysis for Cycle 2".
(7) Letter to D.J. Skovholt from J.S. Abel, dtd 10/18/73, subj: Scram Reactivity Limitations for Dresden Units 2 and 3 and Quad-Cities Units 1 and 2.

3.3 LIMITING CONDITION FOR OPERATION

3. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

    a. Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by
    4 dropout of any increment of any one control blade would not make the
    13 core more than 0.013 Δk super-critical.

    b. Whenever the reactor is in the Startup/Hot Standby or Run mode below 10% rated thermal power, the Rod Worth Minimizer shall be operable. A second operator may be used as a substitute for an inoperable
    13 rod worth minimizer fails after withdrawal of at least twelve control rods. to the fully withdrawn position.

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three
4 counts per second and these SRMs are fully inserted.

4.3 SURVEILLANCE REQUIREMENT

3. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified after loading the sequence.

    Prior to the start of control rod withdrawal towards criticality the capabili of the Rod Worth Minimizer to properly
    4 fulfill its function shall be verified by the following checks:

    a. The RWM computer on line diagnostic test shall be successfully performed.

    b. Proper annunciation of the selection error of one out-of-sequence control rod shall be verified.

    c. The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.

4. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels have an observed count rate of at least three counts per second.

5. During operation with limiting control rod patterns, as determined by the nuclear engineer, either:

   a. Both RBM channels shall be operable; or

   b. Control rod withdrawal shall be blocked; or

   c. The operating power level shall be limited so that the MCHFR will remain above 1.0 assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

13

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	5.00

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power, with
3 reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from
10 the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

(Revised with Change No. 13 issued 3/25/74)

3.3 LIMITING CONDITION FOR OPERATION

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	5.300

13

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

3. If Specification 3.3.C.1 cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shut down immediately upon determination that average scram time is deficient.

10

4. If Specification 3.3.C.2 cannot be met, the deficient control rod shall be considered inoperable, fully inserted into the core, and electrically disarmed.

4.3 SURVEILLANCE REQUIREMENT

2. Following a controlled shutdown of the reactor, but not more frequently than 16 weeks nor less frequently than 32 weeks intervals, 50% of the control rod drives in each quadrant of the reactor core shall be measured for scram times specified in Specification 3.3.C. All control rod drives shall have experienced scram-time
10 measurements each year. Whenever 50% of the control rod drives scram times have been measured, an evaluation shall be made to provide reasonable assurance that proper control rod drives performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the semiannual operating report to the AEC.

3.3 Limiting Condition for Operation Bases (cont.)

3. Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013Δk supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident. These sequences are developed prior to initital operation of the unit following any refueling outage and the requirement that an operator follow these sequences is supervised by the RWM or a second qualified station employe. This 0.013Δ k limit,
13 together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference 1.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2 and 14.2.1.4 of the Safety Analysis Report. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident. These Techniques are described in a topical report[1] and two supplements [2][3].

By using the analytical models described in those reports coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 10% of rated power, the specified limit on in-sequence control rod or control rod segment worths will limit the peak fuel enthalpy to less than 280 cal/gm. Above 10% power even single operator errors
13 cannot result in out-of-sequence control rod worths which are sufficient to reach a peak fuel enthalpy of 280 cal/gm should a postulated control rod drop accident occur.

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(1) Paone, C. J., Stirn, R. C. and Wooley, J.A., "Rod Drop Accident Analysis for Large BWR's", NEDO-10527, 3/72.

(2) Stirn, R.C., Paone, C.J., and Young, R.M., "Rod Drop Accident Analysis for Large BWR's", Supplement 1 - NEDO-10527, 7/72.

(3) Stirn, R.C., Paone, C.J., and Haun, J.M., "Rod Drop Accident Analysis for Large BWR's Addendum No. 2, Exposed Cores", Supplement 2 - NEDO-10527, 1/73.

3.3 Limiting Conditions for Operation Bases (cont.)

The following conservative or worst-case bounding assumptions have been made in the analysis used to determine the specified 0.013 delta k limit on in-sequence control rod or control rod segment worths. The allowable boundary conditions used in the analysis are quantified in reference 4. Each core reload will be analyzed to show conformance to the limiting parameters.

a. A maximum inter-assembly local power peaking factor of 1.3 which is not expected to be reached during future reloads. (5)

b. An end-of-cycle delayed neutron fraction of 0.005.

13 c. A beginning-of-life Doppler reactivity feedback.

d. The rod scram insertion rate shown in Specification 3.3.C.

e. The maximum possible rod drop velocity of 3.11 ft/sec.

f. The design accident and scram reactivity shape function.

g. The moderator temperature at which criticality occurs.

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(4) Exhibit A attached to September 14, 1973 letter from Byron Lee, Commonwealth Edison Company, to J. F. O'Leary, U. S. Atomic Energy Commission.

(5) To include the power spike effect caused by gaps between fuel pellets.

In most cases the worth of in-sequence rods or rod segments will be substantially less than 0.013 delta k. Further, the addition of 0.013 delta k worth of reactivity, as a result of a rod drop and in conjunction with the actual values of the other important accident analysis parameters described above, would most likely result in a peak fuel enthalpy substantially less than 280 cal/gm design limit. However, the 0.013 delta k limit is applied in order to allow room for future reload changes and ease of verification without repetitive Technical Specification changes.

13 Should a control drop accident result in a peak fuel energy content of 280 cal/gm less than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in an offsite dose well below the guideline value of 10 CFR 100. For 8 x 8 fuel, less than 850 rods are conservatively estimated to perforate with nearly the same consequences as for the 7 x 7 fuel case because of the rod power differences.

3.3 Limiting Condition for Operation Bases (cont.)

The Rod Worth Minimizer provides automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Ref. SAR Section 7.9. It serves as a backup to procedural control of control rod worth. In the event that the Rod Worth Minimizer is out of service, when required, a licensed operator or other qualified technical employee can manually fulfill the control rod pattern conformance function of the Rod Worth Minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron levels. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

COMMONWEALTH EDISON COMPANY

DOCKET NOS. 50-254 AND 50-265

INTRODUCTION

Commonwealth Edison Company submitted, by a letter dated November 16, 1973, proposed changes to Section 3.3 of the Technical Specifications and the associated bases in Sections 1.2, 2.1, and 3.3.

References supporting the proposed change were (1) a similar proposal for Dresden Unit 3 submitted by letter dated September 14, 1973, and (2) Dresden Special Report No. 29 dated July 2, 1973.

One proposed change increases the allowable in-sequence control rod worth and results in a less conservative, but justifiable, value for this parameter. The other proposed change reduces control rod scram insertion time which provides a more conservative reactor shutdown rate requirement. We have included additional restrictions for operability requirements of the Rod Worth Minimizer during reactor startup, but allow six months for this change to become effective as it may result in modifications to the equipment.

EVALUATION

The maximum allowable in-sequence incremental reactivity worth of the control rod is based on results of the evaluation of the control rod drop accident. The peak fuel enthalpy for such an accident must be limited to not greater than 280 cal/gm. This limit is established to provide reasonable assurance that the energy release will be sufficiently low to preclude fragmentation of fuel rods during the accident.

The peak fuel clad enthalpy is most sensitive to the following input parameters:

1. Steady state accident reactivity shape function.
2. Total control rod reactivity worth.
3. Maximum inter-assembly local power peaking factor (P_L - normalized over four bundles).
4. Delayed neutron fraction.
5. Scram reactivity shape function.
6. Doppler reactivity feedback.
7. Moderator temperature.

The results of the analysis are summarized in Commonwealth Edison Company's letter dated September 14, 1973, "Proposed Change of the Maximum Allowable In-Sequence Control Rod Worth Permitted by the Technical Specifications for Dresden Unit 3, AEC Docket No. 50-249". We have reviewed this summary and other related documents. The related documents are a General Electric Company topical report, NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors", two supplements to this report and the Commonwealth Edison Company submittal dated May 2, 1973, which contained information on the rod drop analysis for Dresden Units 2 and 3 and Quad-Cities Units 1 and 2.

The analytical methods used by the General Electric Company (GE) to evaluate the consequences of the rod drop accident have been reviewed by the staff and independent calculations have been performed by Brookhaven National Laboratory which show reasonable agreement with GE results. Based on these reviews, it is concluded that the analytical methods used by GE are acceptable.

Application of the GE analytical methods to operating reactors requires that the input parameters conservatively represent the reactor core over a broad range of operating conditions. The proposed changes to the Technical Specifications include, in the Bases, a set of boundary conditions which are used to calculate the maximum allowable reactivity worth of a control rod. It is not expected that these boundary conditions will be exceeded for reactor cores of current design. The boundary conditions include a maximum inter-assembly local power peaking factor, an end-of-cycle delayed neutron fraction, a beginning of life Doppler reactivity feedback, the technical specification control rod scram insertion rate, a control rod drop velocity of 3.11 ft/sec, and specified accident and scram reactivity shape functions. The rod drop velocity of 3.11 ft/sec is based on tests with a "worst case" rod built with maximum clearances and features known to contribute the high rod drop velocities. The difference between the mean rod drop velocity and the 99.9% confidence limit for a group of production rods was added to the mean velocity obtained for the "worst case" control rod. We have included in the Bases the value 0.005 end-of-cycle delayed neutron fraction to further define the boundary assumptions that were used in the calculations. In addition, we have added a statement to the Bases that each reload core must be analyzed to show conformance to the bounding assumptions. The peak fuel enthalpy resulting from an in-sequence rod drop accident within the above boundary conditions is calculated not to exceed 280 cal/gm, which is acceptably below the peak fuel enthalpy at which prompt fuel dispersal would occur based on the SPERT tests. Based on the above, the resultant maximum allowable in-sequence rod worth of 1.3% delta k/k is acceptable.

Separate consideration is being given to the potentially small effect of compaction of boron carbide in the control rods on the rod drop accident in the event of inverted poison tubes. The evaluation of the effect of possible inverted poison tubes on the allowable in-sequence rod worth is currently in progress and if determined necessary, appropriate changes to the allowable control rod reactivity worth will be made.

If a control rod is withdrawn out of sequence, a rod worth of greater than 1.3% delta k/k could result. In the event of rod drop accident associated with such an out-of-sequence rod, the peak fuel enthalpy could exceed 280 cal/gm. The rod worth minimizer (RWM) is designed as an operator aid to prevent an out-of-sequence rod withdrawal. Current Technical Specifications allow the RWM to be bypassed if it is inoperable during a reactor startup provided that a second operator is assigned to monitor the rod withdrawal sequence. To increase the control on RWM availability during reactor startups, the technical specification is being changed to require that the RWM be operable for the withdrawal of a significant number of control rods. The effective date of the change in technical specifications concerning RWM operability is being deferred for six months to allow any necessary upgrading of the RWM to be accomplished.

CONCLUSIONS

Based on the above, we have concluded that this action does not involve a sigificant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered.

/S/

John I. Riesland
Operating Reactors Branch #2
Directorate of Licensing

/S/

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

Date: MAR 25 1974

Docket Nos. 50-254
and 50-265

MAR 2 5 1974

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

Change No. ~~13~~ 14
License Nos. DPR-29
and DPR-30

Gentlemen:

*References:

1. Ltr fm Mr. J. S. Abel to Mr. D. J. Skovholt, dtd 11/14/73
2. Ltr fm Mr. J. S. Abel to Mr. D. J. Skovholt, dtd 09/25/73
3. Ltr fm Mr. L. D. Butterfield to Mr. D. J. Skovholt, dtd 07/03/73
4. Ltr fm Mr. L. D. Butterfield to Mr. D. J. Skovholt, dtd 06/05/73

We have reviewed your request, dated February 19, 1974, for a temporary waiver of one of the limiting conditions of operations concerning the chlorine effluent contained in the Quad Cities Non-Radiological Technical Specifications. Appendix B to License Nos. DPR-29 and DPR-30.

Based on our review of your chlorine test program (References 1-4)* and the surveillance requirements listed below we have concluded that waiving that portion of the Technical Specifications having to do with chlorine effluent until we receive the additional information requested in my letter dated February 12, 1974 is justified. The review of the chlorine test program is necessary in order to establish revised Technical Specifications which are consistent with the protection of the environment. The waiver during the review period does not present a significant hazards considerations, and there is reasonable assurance that the health and safety of the public will not be endangered.

Pursuant to Section 50.59 of 10 CFR Part 50, the requirements of Section 1.1 of Appendix B of the Technical Specifications of Facility Operating License Nos. DPR-29 and DPR-30 are hereby waived until

C/P
RJ

OFFICE ➤						
SURNAME ➤						
DATE ➤						

MAR 2 5 1974

July 1, 1974 provided that the station is operated within the following limitations:

1. The total free and combined chlorine in the discharge bay shall not exceed 1.0 ppm.

2. The period during which chlorination is performed shall not exceed 2 hours a day.

3. If the measured chlorine concentration exceeds the limit in No. 1 above, chlorination shall immediately cease and not be started within 8 hours subsequent to the excess measurement.

4. The Technical Specifications, Appendix B, Specification 2.1, except for the numerical values, shall be the minimum surveillance requirements during this waiver period.

Sincerely,

/s/

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

cc: Mr. B. B. Stephenson, Superintendent
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