APR 21 1975

DISTRIBUTION
Docket
NRC PDR
Local PDR
ORB #2 Reading
Attorney, OELD
OI&E (3)
NDube
BJones (4)
JMMcGough
JSaltzman
RMDiggs
RReid
RMDiggs
DLZiemann
SKari
WOMiller
BScharf (15)
TJCarter
PCollins
SVarga
CHebron
ACRS (14)
JRBuchanan
TBAbernathy

Docket No. 50-265

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 9 to Facility License DPR-30 for the Quad Cities Station Unit 2. This amendment includes Change No. 24 to the Technical Specifications and is in response to your request dated December 13, 1974, and supplements thereto dated December 20, 1974, February 20, 1975, February 27, 1975, and March 27, 1975.

This amendment authorizes operation with 8x8 fuel assemblies, replacement of a relief valve with a relief/safety valve and changes to the reactor vessel relief and safety valve setpoints. However, operation of the reactor is not authorized pending completion of our review of your analysis of emergency core cooling system performance submittal in accordance with the AEC Order for Modification issued December 27, 1974.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann

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

Enclosures:
1. Amendment No. 9 w/Change No. 24
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures: See next page

Notified CE (Hble) at 11:45 A.M. 4/22/75

OFFICE	RL:ORB #2	RL:ORB #2	RL:ORB #2	OELD		RL:AD/ORs	RL:D
SURNAME	RMDiggs	RReid/tc	DLZiemann			NRGoller	AGiambusso
DATE	4/3/75	4/7/75	4/7/75	4/15/75	4/17/75	4/21/75	4/21/75

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

cc w/enclosures:
Mr. Charles Whitmore
President and Chairman
Iowa-Illinois Gas and
Electric Company
206 East Second Avenue
Davenport, Iowa 52801

John W. Rowe, Esquire
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza
Chicago, Illinois 60670

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, N. W.
Washington, D. C. 20036

Moline Public Library
504 - 17th Street
Moline, Illinois 61265

Mr. Robert W. Watts, Chairman
Rock Island County Board of
Supervisors
Rock Island County Courthouse
Rock Island, Illinois 61201

cc w/enclosures and cy of CECO filings
dtd. 2/20/75, 2/27/75 and 3/27/75 & 3/14/75:
Mr. Leroy Stratton
Bureau of Radiological Health
Illinois Department of Public Health
Springfield, Illinois 62706

Mr. Gary Williams
Federal Activities Branch
Environmental Protection Agency
230 South Dearborn Street
Chicago, Illinois 60604

Mr. Ed Vest
Environmental Protection Agency
1735 Baltimore Avenue
Kansas City, Missouri 64108

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 9
License No. DPR-30

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated December 13, 1974, and supplements dated December 20, 1974, February 20 and 27, 1975, and March 27, 1975, comply 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraphs 3.B and 3.C of Facility License No. DPR-30 are hereby amended to read as follows:

OFFICE ➤						
SURNAME ➤						
DATE ➤						

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 24.

C. Restrictions

Operation of the reactor is not authorized beyond the point in the fuel cycle at which the reactivity insertion rate during a scram is less than that of Curve B on Figure 1 of "Dresden Station Special Report 29, Supplement B" dated March 29, 1974.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
A. Giambusso

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 24 to the
Technical Specifications

Date of Issuance: APR 21 1975

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 9

CHANGE NO. 24 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Delete pages 22, 24, 25, 26, 75, 76, 83, 92, 100 and 119 from the Technical Specifications and insert the attached revised pages bearing the same numbers. The changed areas on the revised pages are shown by a marginal line.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature in the run mode which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressures does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram in the run mode assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram in the run mode anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure in the run mode there is no increase in neutron flux.

I. Turbine EHC Control Fluid Low Pressure Scram - The turbine EHC control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the turbine control valve fast closure scram since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the turbine control valve fast closure scram. The scram setpoint of 900 psig is set high enough to provide the necessary anticipatory function and low enough to minimize the number of spurious scrams. Normal operating pressure for this system is 1250 psig. Finally the control valves will not start until the fluid pressure is 600 psig. Therefore, the scram occurs well before valve closure begins. | 24

J. Condenser Low Vacuum Scram - Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure in the run mode. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being

1.2 SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor coolant system pressure shall not exceed 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

A. Reactor coolant high pressure scram shall be $\leq$1060 psig.

B. Primary system safety valve Nominal settings shall be as follows:

1 valve at 1125 psig*
2 valves at 1240 psig
2 valves at 1250 psig
4 valves at 1260 psig

24

The allowable setpoint error for each valve shall be $\pm$1%.

*Target Rock combination safety/relief valve

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 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 strenght are comparable for the primary system piping snd 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[4-6]. 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 limit the reactor pressure to a value 24

---

(4) SAR, Section 11.2.2.

(5) "Quad Cities 1 Nuclear Power Station First Reload License Submittal," Section 6.2.4.2, February 1974. 24

(6) NEDO-20693, "General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Quad Cities Nuclear Power Station Unit 2, December 1974.

1.2 Safety Limit Bases (Cont'd)

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full scale pressure recorder.

2.2 Limiting Safety System Setting Bases

In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the high pressure scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus exceeding the pressure safety limit. The pressure scram is actually a backup protection to the high flux scram which was analyzed[9] in Section 4.4.3 of the SAR, and reexamined for Unit 1 fuel cycle 2 in "Dresden Station Special Report No. 29 Supplement B." If the high flux scram were to fail during a maximum pressure transient (also assuming failure of the turbine stop valve closure scram, failure of the bypass system to actuate and failure of the relief valves to open), the pressure would rise rapidly due to void reduction in the core. A high pressure scram would occur at 1060 psig.

Unit 1

The pressure at the botton of the vessel is about 1163 psig when the first safety valve opens and about 1290 psig when the last valve opens. Both valves are clearly within code requirements. Vessel dome pressure reaches less than 1277 psig with a peak at the bottom of vessel less than 1301 psig. Therefore, the pressure scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

(9) "Quad Cities 1 Nuclear Power Station First Reload License Submittal", Section 6.2.4.2, February 1974.

Unit 2

The pressure at the botton of the vessel is about 1158 psig when the first safety valve opens and about 1314 psig when the last valve opens. Both values are clearly within the code requirements.

Vessel dome pressure reaches about 1292 psig with the peak at the bottom of the vessel near 1322 psig. Therefore, the neutron flux scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig. 24

3.3 LIMITING CONDITION FOR OPERATION

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:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

24

4.3 SURVEILLANCE REQUIREMENT

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 reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

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	2.12
90	3.80

24

2. The maximum scram insertion time for 90% insertion of any operable control rods 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.

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 week 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 test measurements each year. Whenever all 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 NRC.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. Section 7.4.5.3 of the SAR references the effects of a rod block monitor failure. These amendments show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCHFR's less than 1.0. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCHFR from becoming less than 1.0. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCHFR remains greater than 1.0. Reference (1) shows the control rod scram reactivity used in analyzing the transients. Reference (1) should not be confused with the total control rod worth, 18%Δk, as listed in some amendments to the SAR. The 18%Δk value represents the amount of reactivity available for withdrawal in the cold clean core, whereas the control rod worths shown in Reference (1) represent the amount of reactivity available for insertion (scram) in the hot operating core. The minimum amount of reactivity to be inserted during a scram is controlled by permitting no more than 10% of the operable rods to have long scram times. In the analytical treatment of the transients, 390 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 270 milliseconds. Approximately 70 milliseconds after neutron flux reaches the

24

24

(1) "Dresden Station Special Report No. 29, Supplement B," Figure 1.

24

3.4 Limiting Condition for Operation Bases

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of 600 ppm of boron in the reactor core in approximately 90 to 120 minutes with imperfect mixing. A boron concentration of 600 ppm in the reactor core is required to
24 bring the reactor from full power to a 3%Δk subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional margin of 150 ppm in the reactor core for imperfect mixing of the chemical solution in the reactor water. A normal quantity of 3470 gallons of solution having a 13.4% sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (90 to 120 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required pumping rate of 39 gallons per minute, the maximum storage volume of the boron solution is established as 4,875 gallons (195 gallons are contained below the pump suction and, therefore, cannot be inserted).

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Experience with pump operability indicates that monthly testing is adequate to detect if failures have occurred.

The only practical time to test the standby liquid control system is during a refueling outage and by initiation from local stations. Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of less than once each refueling outage unnecessary. A test of explosive charges from one manufacturing batch is made to assure that the charges are satisfactory. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

B. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the remaining system will perform its intended function and that the reliability of the system is good is obtained by demonstrating operation of the pump in the operable circuit at least once daily. A reliability analysis indicates that the plant can be operated safely in this manner for seven days.

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT
D. Automatic Pressure Relief Subsystems	D. Automatic Pressure Relief Subsystems
1. The Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig, irradiated fuel is in the reactor vessel and prior to reactor startup from a Cold Condition.	Surveillance of the automatic pressure relief subsystems shall be performed as follows: 1. During each operating cycle the following shall be performed: a. A simulated automatic initiation which opens all pilot valves, and b. With the reactor at low pressure each relief valve shall be manually opened until thermocouples downstream of the valve indicate fluid is flowing from the valve. c. A logic system functional test shall be performed each refueling outage.
17 & 24 2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, continued reactor operation is permissible only during the succeeding thirty days unless repairs are made and provided that during such time the HPCI subsystem is operable.	2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter.

3.6 LIMITING CONDITION FOR OPERATION

E. Safety and Relief Valves

1. Prior to reactor startup for power operation and during reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320°F, all nine of the safety valves shall be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

2. If Specification 3.6.E.1 is not met, the reactor shall remain shutdown until the condition is corrected or, if in operation, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be below 90 psig and 320°F within 24 hours.

4.6 SURVEILLANCE REQUIREMENT

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outage. The popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (psig)
1	1125*
2	1240
2	1250
4	1260

24

The allowable set point error for each valve is ±1%.

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Number of Valves	Set Point (psig)
1	1125*
2	≤ 1130
2	≤ 1135

24

*Target Rock combination safety/relief valve.

24

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 9 TO FACILITY OPERATING LICENSE NO. DPR-30

(CHANGE NO. 24 TO THE TECHNICAL SPECIFICATIONS)

COMMONWEALTH EDISON COMPANY AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES UNIT 2

DOCKET NO. 50-265

INTRODUCTION

By application dated December 13, 1974, Commonwealth Edison Company (CE) requested authorization to operate Quad Cities Unit 2 with reload fuel assemblies in the core. According to CE's plan, as many as 148 8x8 reload fuel assemblies will replace an equal number of fuel assemblies presently in the core. In conjunction with the reload, it has been proposed that several areas of the Technical Specifications be changed and that certain modifications to the reactor be approved.

The following Technical Specification changes and plant modifications have been proposed:

1. Reload with 8x8 fuel,

2. Replacement of a relief valve with a relief/safety valve and change of relief and safety valve setpoints,

3. Change of average power range monitor (APRM) and rod block monitor (RBM) trip settings to be discussed in the safety evaluation related to item 4 below,

4. Adoption of Technical Specification based on the new GEXL boiling transition correlation and

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

5. Adoption of Emergency Core Cooling System Technical Specifications based on a reanalysis required by AEC Order for Modification issued December 27, 1974 (applicable to Quad Cities 2 only).

To expedite the review process some of the proposed changes will be reviewed in different safety evaluations. Items 1 and 2 are discussed in this safety evaluation and items 3, 4, and 5 will be evaluated in a subsequent safety evaluation to be issued in connection with this reload authorization.

EVALUATION

1. Reload With 8x8 Fuel (Quad Cities Unit 2 Only)

The application to reload Quad Cities Unit 2 with 8x8 fuel assemblies proposes appropriate Technical Specifications required to load and operate with 8x8 fuel assemblies. Supplements to the application were submitted by letters dated December 20, 1974, February 20, 1975, February 27, 1975, and March 27, 1975. The acceptability of the neutronic, thermal-hydraulic, and mechanical design of the 8x8 fuel assemblies during normal operation, operational transients and postulated accidents was evaluated by the NRC staff in a previous report (1). The use of 8x8 fuel assemblies for reloads was also reviewed by the Advisory Committee on Reactor Safeguards and discussed in its report dated February 12, 1974 (2). The use of 8x8 reload fuel assemblies in the Quad Cities Unit 1 (which is essentially identical to the Quad Cities Unit 2 reactor) was evaluated and approved by Amendment No. 10 to the Facility Operating License No. DPR-29 dated June 5, 1974.

With one exception, the evaluations of the acceptability of the reload fuel for the Quad Cities Unit 1 core are applicable to the Quad Cities Unit 2 reload fuel. A principle design change for this reload 8x8 fuel is the use of leaf springs to minimize the bypass flow area between the fuel assembly shroud and the lower end fitting. The effect of this design change is discussed below.

Our safety evaluation of this reload (Reload No. 1) for the Quad Cities Unit 2 core is based on the licensee's application as amended, and on information contained in a GE topical report, NEDO-20360 (3) referred to in the application. The NEDO-20360 report is still being evaluated

(1) Technical Report on the General Electric Company 8x8 Fuel Assembly, dated February 5, 1974, by the Directorate of Licensing

(2) Report on General Electric 8x8 Fuel Design for Reload Use, Advisory Committee on Reactor Safeguards, February 12, 1974.

(3) General Electric Boiling Water Reactor General Reload Application for 8x8 Fuel, NEDO-20360 Revision 1 (November 1974).

by the staff for use as a topical. Our use of that report in this analysis was limited to considerations applicable to Quad Cities Unit 2 and does not imply acceptability of its use for other facilities.

Authority to load but not operate with Reload 1 fuel was approved by Amendment No. 8, with Change No. 23, to Facility Operating License No. DPR-30 dated February 13, 1975.

The reference Quad Cities Unit 2 Reload 1 consists of 576 initial 7x7 fuel assemblies and one hundred forty-eight 8x8 reload fuel assemblies which are scatter loaded throughout the interior of the core. Four fuel assemblies surrounding a control blade will contain only one 8x8 reload fuel assembly. This loading scheme assures that, in the core interior, the higher enrichment 8x8 reload fuel assembly will be "paired" with three lower powered exposed 7x7 fuel assemblies. No significant fuel loading asymmetries will exist.

The NRC staff has previously reviewed[1] the mechanical design of the 8 x 8 reload fuel assemblies. In addition the 8 x 8 reload fuel assemblies for Quad Cities Unit 2 are of similar design to the 8 x 8 fuel assemblies approved for use in Quad Cities Unit 1 by Amendment No. 10 to Facility Operating License No. DPR-29. Because Quad Cities Units 1 and 2 operate at identical conditions and the fuels used are nearly identical, the evaluation of mechanical design discussed in our safety evaluation for Amendment No. 10 to Facility Operating License No. DPR-29 is applicable to the 8x8 reload fuel assemblies for Quad Cities Unit 2. The 8x8 fuel assemblies for Quad Cities Unit 2 are of similar design and material to the 7x7 fuel assemblies which have successfully been operated at Quad Cities Unit 2. Both the 8x8 and 7x7 fuel assemblies will operate at the same pressure and temperature and the fluid velocity and quality will be nearly identical and, therefore, the new 8x8 fuel assemblies are expected to exhibit the same operational characteristics as the previously operated 7x7 fuel assemblies.

Mechanical and operating parameters for the 8x8 assemblies are compared with the 7x7 assemblies in Table 1. The smaller diameter rods, with lower linear heat generation rate and larger cladding thickness/diameter ratio for the 8x8 fuel design, result in increased engineering safety margins when compared with the 7x7 fuel assemblies. The maximum design linear power is significantly reduced for the 8x8 fuel. The reload fuel incorporates finger springs for controlling moderator/coolant bypass flow at the interface

TABLE 1

COMPARISON OF PARAMETERS FOR 8X8 AND 7X7

ROD FUEL ASSEMBLY DESIGN

	7X7	8X8
Pellet Outside Diameter (in.)	0.477	0.416
Rod Outside Diameter (in.)	0.563	0.493
Rod-to-Rod Pitch (in.)	0.738	0.640
Water-Fuel Ratio (cold)	2.53	2.60
U Bundle Weight (pounds)	412.8	404.6
Cladding Thickness (mils)	37	34
Active Fuel Length (in.)	144	144
Maximum Design Power (kw/ft)	17.5	13.4

of the channel and fuel bundle lower tie plate. This device has performed satisfactorily in other reactors on more than 900 assemblies that have been inspected.

Four different U-235 enrichments are used in the reload fuel assemblies to reduce the local power peaking factor. In addition, gadolinia-urania pellets are used in four of the highest enrichment rods. Gadolina is a burnable poison and supplements the control rods in flattening the power distribution of the core. The use and performance of gadolinia - bearing fuel was evaluated by us for the initial core of Quad Cities 1 and 2 and found acceptable.

Fuel performance calculations that account for the effects of fuel densification have been performed with our approved version of the General Electric analytical model, GEGAP III[4,5] . The effects of fuel densification on the fuel rod will be that of increasing the stored energy, increasing the linear thermal output, and increasing the probability of local power spike from axial gaps. The primary effects of densification on the fuel rod mechanical design are manifested in calculations on fuel-clad gap conductance and cladding collapse time. The approved analytical model incorporates time-dependent fuel densification, time-dependent gap closure and cladding creepdown for the calculation of gap conductance. Clad collapse has not been observed in BWR fuel rods and is calculated to occur at core residence times in excess of 5 years. The fuel residence time is expected to be approximately 4 years. Therefore, densification effects have been appropriately considered.

Accident induced loads and stresses have been calculated for both the 7x7 and 8x8 fuel assemblies using the same methods. The limiting accident loads result from a steam line break. The pressure difference following a steam line break are less than 10% greater than normal operating pressure differences. As in normal operation, the pressure differences in an 8x8 assembly following a steam line break are 5 to 10% greater than in a 7x7 assembly. The loads following a steam line break are well below the allowable loads.

The reload 8x8 fuel design is currently in operation in Nine Mile Point, Pilgrim 1, Monticello, Dresden 2 and 3, Quad Cities 1 and Vermont Yankee. Post irradiation examination of these assemblies

---

(4) GEGAP-III, "A Model for the Prediction of Pellet Clad Thermal Conductance in BWR Fuel Pools," NEDO-20181, December 3, 1973, Supplement 1 (Proprietary)

(5) V. A. Moore, NRC Letter to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket No. 50-321, March 22, 1974

OFFICE➤						
SURNAME➤						
DATE➤						

at Monticello after one complete cycle indicated satisfactory performance. On the basis of our review of the generic reload report (3) and the confirmatory results from irradiated assemblies, we conclude that for Quad Cities 2 reload: (1) the fuel rod mechanical design provides acceptable engineering safety margins for normal operation; (2) the effects of fuel densification have been acceptably accounted for in the fuel design, and (3) fuel rod and channel box integrity during transients and accidents will perform satisfactorily.

We have also reviewed the nuclear design of the reload fuel. The CE submittal indicates that the nuclear characteristics of the Reload 1 fuel assemblies are similar to those previously loaded. However, a reassessment of the dynamic void coefficient has been made resulting in the adoption of a coefficient about 20 percent more negative than that used in the first fuel cycle.

Those events which would be affected by using a more negative void coefficient have been reanalyzed. Our evaluation of the results are presented below with the discussion of anticipated transients. In other respects, the reactivity coefficients and total control system worths of the reconstituted core will not differ significantly from those values which were reported previously for Quad Cities Unit 2. In addition, the nuclear characteristics of the Quad Cities Unit 2, Cycle 2 with Reload 1 fuel assemblies are quite similar to the previously approved Reload 1 fuel assemblies for Quad Cities Unit 1.

The application also indicates that the shutdown margin of the reconstituted core meets the technical specification requirement that the core be at least 0.25% Δk subcritical in the most reactive operating state with the control rod of largest reactivity worth fully withdrawn and with all other control rods fully inserted. The report predicts that, at a core average exposure of 10,200 MWD/T at the end of Cycle 2, the shutdown margin will be 3.49% Δk with largest worth control rod fully withdrawn and all other rods fully inserted. The analysis applies to control rods whose blades do not contain inverted boron filled tubes. The control rods of Quad Cities Unit 2 were examined during the current refueling outage, and it has been determined that those control rods remaining in the reactor with inverted boron filled tubes would have a negligible effect on the shutdown margin analysis.

The application states, and we agree, that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by 0.03 Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design is met.

The basic criterion for the storage of fuel for Quad Cities Unit 2 is that k_{eff}, of the fuel as stored in the fuel pool is ≤ 0.90. This is achieved if the uncontrolled infinite multiplication factor of a single fuel assembly is less than 1.26 at 65°C. The 8x8 reload fuel at both zero exposure and the peak reactivity point has an infinite multiplication factor of <1.26, and therefore, meets the fuel storage requirements for Quad Cities 2.

Based on our evaluation as reported[1], we conclude that a mixed 8x8 and 7x7 core will be nearly identical, neutronically, to a 7x7 core and that the nuclear design is acceptable.

Thermal-hydraulic methods used to analyze flow rates are discussed in Reference 3. These methods are the same as those used previously to analyze reactor conditions and are acceptable. To provide adequate thermal margin during normal steady-state operation, the reactor is limited to operating with maximum LHGR's of 17.5 Kw/ft for 7x7 fuel and 13.4 Kw/ft for 8x8 fuel. General Electric has predicted an increase in bypass flow caused by channel wall deflections. The deflection model was developed from measurements of creep deformation of the shroud at operating conditions. To nullify this potential increase in flow area, leaf springs have been attached to each of the four sides of the lower end fittings of the reload fuel. The effect of this change on bypass flow and the different hydraulic characteristics of the 8x8 fuel assemblies are accounted for in the steady-state and transient analyses that are presented.

Based on a review of the information provided by the licensee, we conclude that:

1. The linear heat generation rate design criteria and analysis methods, which are the same as those used previously for justifying plant safety, are acceptable.

2. The licensee has accounted for the different hydraulic characteristics of the reload fuel in an acceptable manner.

Abnormal operational transients were discussed in the staff report for 8x8 reload fuel[1]. As previously discussed, the mechanical, nuclear, and thermal hydraulic characteristics of the 7x7 and 8x8 fuel are similar and will respond to the transients similarly.

The application and supplements include analyses of the events which approach thermal limits. Previously these events were analyzed to determine the margin to the minimum critical heat flux. In conjunction

with this reload, it has been proposed that the basis for the thermal limits be changed from the Hench-Levy critical heat flux correlation previously used to the GEXL transition boiling correlation recently developed by the General Electric Company. Our review of the application of the GEXL correlation to normal operation and the approach to thermal limits during anticipated transients will be discussed in a separate safety evaluation.

There are two events, however, the results of which are affected by changes occurring in conjunction with the reload but which are not primarily affected by the change in the bases for thermal limits. The two events are a turbine trip assuming failure at the bypass system and main steam line isolation valve closure assuming failure of the valve position scram. These events are discussed below.

For Quad Cities Unit 2 fuel cycle 2, the scram reactivity curves at the end of the cycle will change such that the reactivity insertion rate will be slower than that at the end of the first fuel cycle. The change in scram reactivity insertion rate is the same as that previously evaluated for Quad Cities 1[6]. The change in scram reactivity insertion rate results in an increase in the peak pressure during pressurization events. The analysis of a turbine trip, assuming failure of the bypass system, is used to evaluate the adequacy of the relief valve system capacity. The analysis of turbine trip without bypass has shown that to maintain acceptable peak pressure margins, reduction in power level or plant modifications are necessary. Plant modifications have been proposed which reduce but do not eliminate the power level restrictions needed to maintain acceptable peak pressure margins. The modifications are discussed in section 2 of this evaluation. The more negative dynamic void coefficient discussed above results in a greater reactivity increase due to void collapse during overpressure events thereby increasing the power surge and the peak pressure. The transient following turbine trip with failure of bypass has been reanalyzed with the assumption that the plant modifications have been completed and with the more negative void coefficient. The results show that acceptable peak pressure margins are maintained for plant operation at 100 percent of rated power until the scram reactivity decreases to below the generic B curve which has been calculated to occur at a fuel exposure increase of about 3200 MWD/T. CE has concluded[7]

(6) Dresden Station Report No. 29, Supplement B, Transient Analyses for Dresden 3 Cycle 3 and Quad Cities 1 Cycle 2. (March 29, 1974)

(7) "Quad Cities Station Unit 2 Reload 1 Licensing Submittal Supplement C," dated March 27, 1975.

that additional analyses are required to define acceptable power level limits with less than rated core flow rates during that portion of the fuel cycle that the scram reactivity insertion rates are less than that of the generic B curve. CE has stated that such analyses will be provided before that point in the fuel cycle that the analyses are required.

Therefore, the operation of Quad Cities Unit 2 will not be authorized as a license condition when the scram reactivity insertion rate is less than that of the generic B curve as presented in Reference 6.

The margin to thermal limits for the turbine trip without bypass event will be discussed in the safety evaluation concerning the GEXL correlation.

The effect of the change in reactivity insertion rate and dynamic void coefficient on the safety valve sizing event, main steam isolation valve closure with a delayed scram, has been evaluated incorporating the proposed revised safety valve settings. The analysis was performed with the initial power of 100 percent of rated power, using the end-of-cycle reactivity insertion rate ("C" curve), no credit for relief valves, and with shutdown initiated by the high flux trip. The peak reactor vessel pressure for this event was calculated to be 1322 psig which is 53 psi below the allowable peak vessel pressure of 1375 psig and therefore is acceptable.

We conclude that, with the restrictions discussed above, the over-pressure protection during fuel cycle 2 for Quad Cities Unit 2 is acceptable.

The accident review for the control rod drop accident, the refueling accident and the steam line break accident were discussed in our evaluation for the Quad Cities 1 reload for Technical Specification Change No. 19 to Facility Operating License No. DPR-29. Because of the similarity of the reactors and the reload fuel, the Quad Cities 1 evaluation for these accidents is applicable to Quad Cities 2. A reanalysis of ECCS performance and proposed technical specifications for Quad Cities Unit 2 have been submitted in accordance with the AEC Order for Modification dated December 27, 1974. Our evaluation of this analysis and proposed technical specifications will be the subject of a separate safety evaluation to be issued prior to authorizing resumption of operation from the current refueling outage.

Therefore, based upon the above, we conclude that use of 8x8 fuel as reload fuel for Quad Cities Unit 2 is acceptable.

2. Replacement of Relief Valve with a Relief/Safety Valve and Change of Relief and Safety Valve Setpoints

In the Quad Cities Unit 2 reload submittal dated December 20, 1974, CE proposed changes in the Technical Specifications for Quad Cities Unit 2 to allow operation with a combination relief/safety valve in place of an electromatic relief valve. This change was previously authorized for Quad Cities Unit 1 by Amendment No. 8 to Facility Operating License No. DPR-29, issued May 24, 1974. The changes to Quad Cities Unit 2 and the supporting analyses are identical to those for Quad Cities Unit 1 except for the incorporation of a revised dynamic void coefficient as discussed in section 1 of this evaluation.

The changes to the Technical Specifications include requirements for the modified valve, increased pressure setpoints for the spring-loaded safety valves, and more rapid scram times for the control rods. The purpose of the modification and changes is to maintain a 25 psi margin between the calculated peak pressure and the lowest setting of the spring-loaded safety valves for the limiting anticipated pressurization transient, a turbine trip with failure of the turbine bypass valves. The need for the change results from the decrease in rate of negative reactivity insertion during a reactor scram as core average exposure increases. Since Quad Cities Units 1 and 2 are of identical design in features relevant to the evaluation and since the proposed change is identical to the Quad Cities 1 change, the analyses by the licensee and the evaluation by the NRC staff for the two reloads are nearly identical. The only differences are that the use of the revised dynamic void coefficient necessitates reduction of reactor power to 90 percent of rated rather than 93 percent of rated and that the core exposure at which the power reduction will start has been calculated to be 3200 MWD/T in NEDO-20693 rather than 4000 MWD/T calculated for Quad Cities Unit 1.(6) The staff evaluation for this change at Quad Cities Unit 1 is enclosed (dated June 5, 1974).

Coincident with the above change, the bases for the fuel damage safety limit is being revised to adopt the General Electric transition boiling correlation, GEXL. Our evaluation of the margin to the fuel damage limit during a turbine trip with failure of bypass transient will be discussed in a separate safety evaluation concerning use of the GEXL correlation as a basis for Technical Specifications.

Based on the above and subject to the conditions stated above, we conclude that the changeout of one electromatic relief valve to a relief/safety valve and the associated technical specification changes related to relief and safety valve settings and operability are acceptable.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: APR 21 1975

OFFICE➤						
SURNAME➤						
DATE➤						

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-265

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 9 to Facility Operating License No. DPR-30 issued to Commonwealth Edison Company (acting for itself and on behalf of the Iowa-Illinois Gas and Electric Company) which revised Technical Specifications for operation of the Quad Cities Nuclear Power Station Unit 2 located in Rock Island County, Illinois. The amendment is effective as of its date of issuance.

The amendment authorizes operation of Quad Cities Unit 2 using 8x8 fuel assemblies, replacement of a relief valve with a relief/safety valve, and changes to the reactor vessel relief and safety valve setpoints in accordance with the application by Commonwealth Edison Company dated December 13, 1974, as supplemented.

The application for the amendment 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. No request for a hearing or petition for leave to intervene was filed following the Notice of Proposed Issuance of Amendment published in the Federal Register on January 7, 1975 (40 F.R. 1291).

OFFICE ➤						
SURNAME ➤						
DATE ➤						

For further details with respect to this action, see (1) the application for amendment dated December 13, 1974, and supplements thereto dated December 20, 1974, February 20 and 27, 1975, and March 27, 1975, (2) Amendment No. 9 to License No. DPR-30, with Change No. 24, and (3) the Commission's concurrently issued related Safety Evaluation. 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 Moline Public Library at 504 - 17th Street, Moline, Illinois 61265. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 21st day of April 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

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

OFFICE ➤						
SURNAME ➤						
DATE ➤						

~~Enclosure 9~~ FOR AG

CHECKLIST FOR ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

APPLICANT Commonwealth Edison DOCKET NO. 50-265

FACILITY Quad-Cities Unit 2 (Amend. 9 - 848 oper.)

PROJECT MANAGER Bob Reid (7403)

LICENSING ASSISTANT Reba Diggs (7403)

	DATE
Notice of Proposed Issuance Published In FEDERAL REGISTER	January 7, 1975
Action Date	~~After 2/6/75~~
Issuance Package: ELD Concurrence	
1. License Amendment	4/21/75
2. FEDERAL REGISTER Notice	"
3. Staff Evaluation	"
4. Letter to applicant	"
NEPA Determination: ~~Required~~/Not Required	Not Required per R. Reid
For Amendments Affecting Power Level:	
IE Notification and/or Concurrence	N/A
OAI Notification and/or Concurrence 1/	N/A
ADM Ofc. Notification and/or Concurrence	N/A
PA Notification	Not interested per C. Miles 4/8/75.
TR Concurrence	4/17/75

1/ or name change, transfer of facility ownership

~~FEB 9 1976~~